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Evolutionary water cooled reactors: Strategic issues, technologies and economic viability

Proceedings of a symposium held in Seoul, 30 November – 4 December 1998

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FOREWORD

To ensure that nuclear power remains a viable option in the near and medium term, considerable development has been, and is being carried out in several countries on evolutionary water cooled reactor designs. Utility requirements have been formulated to guide these activities by incorporating the large base of experience from current plants, as well as results of research and development programmes, for example, on new safety systems. Common goals of the evolutionary designs are high availability, good operating features, competitive economics and compliance with stringent, internationally recognized, safety objectives.

The International Symposium on Evolutionary Water Cooled Reactors: Strategic Issues, Technologies and Economic Viability was hosted by the Korea Electric Power Corporation (KEPCO) on behalf of the Republic of Korea, in Seoul, Republic of Korea, from 30 November to 4 December 1998. It was organized by the IAEA in co-operation with the OECD Nuclear Energy Agency, the Uranium Institute, the Korean Nuclear Society and the Korea Atomic Industrial Forum.

The symposium reviewed the technological advancements of evolutionary water cooled reactors and their readiness to contribute to the world's near and medium term energy needs. Topics addressed included strategic issues (global energy outlook, the role of nuclear power in sustainable energy strategies, power generation costs, financing, social–political factors, safety requirements); technological advances (instrumentation and control, means of improving prevention and mitigation of severe accidents, development of passive safety systems); and keys to economic viability (simplification, standardization, advancement in construction and project management, and efficient and effective management of plant operations).

The symposium highlighted the importance of continued international co-operation in the development and application of nuclear power for peaceful uses throughout the world. Challenges facing nuclear power in the future include:

- achieving the highest level of safe operation of current plants,
- implementing high level waste disposal,
- establishing a sound basis for defining the potential of nuclear power to contribute to sustainable development,
- achieving further technological advancement that future nuclear plants will be economically competitive with fossil alternatives, especially in deregulated and privatised electricity markets, and
- developing economical small and medium sized reactor designs to provide the nuclear power option to developing countries which have small electricity grids, and also for non-electric applications such as seawater desalination.

Importantly, if these challenges can be met, nuclear power units with stable and low operating and fuel costs have an opportunity to increasingly contribute to the electricity needs of many Member States by providing base load electric power via extended grid networks, and also to provide a clean energy source for non-electric applications.

The IAEA extends its thanks to KEPCO for accomplishing an excellent arrangement of the symposium. The responsible IAEA officers were J. Cleveland, B.O. Cho and R. Lyon of the Division of Nuclear Power.

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SUMMARY

The Symposium on Evolutionary Water Cooled Reactors: Strategic Issues, Technologies and Economic Viability was intended for managers in utilities, reactor design organizations and hardware manufacturing companies, and of government decision makers who need to understand technological advances and the potential of evolutionary water cooled reactors to contribute to near and medium term energy needs.

The topics addressed included:

- strategic issues (global energy outlook, the role of nuclear power in sustainable energy strategies, power generation costs, financing of nuclear plant projects, social-political factors and nuclear safety requirements);
- technological advances (instrumentation and control, means of improving prevention and mitigation of severe accidents, development of passive safety systems);
- keys to economic viability (simplification, standardization, advances in construction and project management, feedback of experience from utilities into new designs, and effective management of plant operations).

Opening addresses were given by:

- Chang-Hee Kang, Minister of Science and Technology of the Republic of Korea;
- M. ElBaradei, Director General of the IAEA; and
- Young-Sik Jang, President and Chief Executive Officer, KEPCO.

Papers were presented in five sessions:

I. Keynote Addresses

II. Key Factors in the Decision-Making Process

III. Advances in Technologies Related to Nuclear Safety

IV. Key Developments of Evolutionary Designs

V. Keys to Economic Viability of Evolutionary Water Cooled Reactors.

All papers were invited, and many were of multiple authorship from different countries or international organizations, reflecting significant international co-operation in their preparation. Submitted posters were presented in two poster sessions. The symposium ended with a session addressing key issues, including the economic challenges, safety objectives for evolutionary designs, the role of nuclear power in sustainable development, national infrastructure and financing strategies important for the wider development of evolutionary plants, and the role of international organizations.

Approximately 320 participants from 28 Member States and three international organizations attended the symposium.

Global energy outlook, and the status and prospects for nuclear power

Global energy demand is growing due to industrialization, economic development and increases in world population. It is projected to almost triple in the mid-21st century. In developing countries in the next thirty years, energy demand is projected to increase two to three-fold, depending on the economic growth scenario. It is anticipated that most of the world's increase in nuclear capacity will be in Asia. For example, the Republic of Korea's nuclear capacity is expected to increase from the current level of 12 GW(e) to 16.7 GW(e) in 2004 and to 24.7 GW(e) in 2010. Japan's nuclear capacity, which currently is 43.9 GW(e), is expected to increase to 52.3 GW(e) by the year 2004 and to 70.5 GW(e) by 2010. China, which currently has 2.2 GW(e) of nuclear capacity, plans to develop an additional 18 GW(e) by the year 2010.

At the beginning of 1998, 437 nuclear reactors operating in 31 countries provided 16.3% of global electricity. Accumulated operating experience with nuclear power plants reached approximately 8500 reactor-years. Of the 437 nuclear power plants in operation, 346 were light water reactors (LWRs) totaling 306 GW(e) and 30 were heavy water reactors (HWRs) totaling 16.4 GW(e). Of the 36 nuclear plants under construction, 25 were LWRs, totaling 21 GW(e), and 8 were HWRs, totaling 3.5 GW(e). Water cooled reactors are a mature and proven technology with an experience base of over 6100 reactor-years for LWRs and over 600 reactor-years for HWRs.

The choice of energy strategies is a national decision that weighs the specific situations and requirements of individual countries. The Korean Government will continue to rely on nuclear power in the future as a main energy supply source because of environmental concerns and the lack of natural resources. For many countries, like the Republic of Korea, nuclear power has advantages for the national security of energy supply, with uranium and thorium providing the resource base.

Concern about the impact of human activities on global climate is growing. Agreements by industrialized countries to lower greenhouse gas emissions were made at the Kyoto Conference in December 1997. At the Fourth Session of the UN Framework Convention on Climate Change Conference of the Parties, convened in Buenos Aires in November 1998, a two year plan of action was adopted which establishes deadlines for finalizing the outstanding details for implementation of the 1997 Kyoto Protocol.

Energy produced from fossil fuels accounts for about half of all man-made greenhouse gas emissions. Except for nuclear and hydropower (which has limited growth potential), there are not yet any other economically viable, minimal-greenhouse gas-emission options for base load electricity generation.

Countries that have significant nuclear power and hydroelectric capacity have markedly lower CO_2 emissions per unit of energy produced than countries with a high fossil fuel component. Today, nuclear power and hydroelectric power each avoid some 8% of global CO_2 emissions annually from energy production. However, unless there is a major thrust by governments to create incentives or to levy heavy carbon taxes, the use of fossil fuels will continue to increase and there will be a major increase in carbon dioxide emissions globally.

Environmental, developmental and national security considerations suggest that nuclear power, together with improved energy efficiency and greater use of renewable energy sources, should continue to be a major component of many national energy strategies. However, in many countries there are substantial hurdles in terms of public and political acceptability and economic competitiveness.

The sharp fall in the price of natural gas, its increased availability, and highly efficient technologies (the combined cycle gas turbine) introduce new competitive elements into power generation choices where natural gas is readily available. Advances in other non-nuclear technologies for power generation (renewables, advanced coal-burning technologies) might also have a major impact on the competitive economics of power generation in the future. A concentrated effort to reduce the capital cost of new nuclear plants is needed to assure that nuclear power will be competitive with alternative options.

The competitiveness of nuclear power would significantly increase if externalities - the considerable indirect and external environmental costs of energy generation - were included under more stringent environmental policies. Indirect costs, such as those for waste management and decommissioning are already included in nuclear power generation costs; these costs are not fully included and are significant for fossil fuels.

Worldwide, there are proven reserves of coal sufficient for at least the next two centuries, and proven reserves of natural gas and oil for several decades at current levels of use. The limits vary significantly from country to country. Improved recovery techniques and oil-shale and tar-sand processing may be capable of doubling the oil and gas resource base. Known uranium reserves assure a sufficient fuel supply for at least 50 years at current levels of use, with reactors operating primarily on a once through cycle without reprocessing spent fuel. In the longer term, introduction of fast breeder reactors would increase the energy potential of today's known uranium reserves by up to 70 times, enough for more than 3000 years at today's usage. Therefore, if the challenges facing nuclear energy can be met, it offers a very long term and sustainable source of energy.

Key factors influencing decisions to build new plants

Key factors in the decision to build evolutionary water cooled reactors include their technological readiness, economic competitiveness, financial arrangements, and social-political factors. Countries with nuclear programmes must focus on the technical and institutional infrastructure that ensures a viable nuclear option, and countries planning to embark on new nuclear programmes must establish this institutional infrastructure.

The large base of experience with water cooled reactors is being used to guide design and development by incorporation into user requirements documents (URDs), such as the Electric Power Research Institute URD and the European Utility Requirements. Technological progress continues, and there is no doubt that evolutionary water cooled reactors will offer improved performance, as can be seen from the steady improvements in performance achieved in current plants. The average energy availability factor for nuclear plants has increased from approximately 70 per cent in 1989 to 77.4 per cent in 1997, with some utilities achieving significantly higher values. This is achieved through integrated programmes covering personnel training, quality assurance, maintenance and inspection, and technological advances in plant components and systems. International co-operation plays a key role. The various programmes of the World Association of Nuclear Operators (WANO) to exchange operating experience, and the activities of the IAEA including projects in nuclear power plant performance assessment and feedback, effective quality management, and information exchange on technology advances, are important examples of international co-operation to improve plant performance.

Improved performance at current plants is supported by better application of existing technologies, such as technologies for processing information on the condition of components, and for performing surveillance and diagnostics. Examples of new technologies improving performance include: high burnup fuel which supports longer cycle length; computer-aided systems to provide early indication of sensor or component degradation; simpler systems for control of hydrogen (systems that require considerably less testing and maintenance and thereby reduce outages); and new materials with superior corrosion resistance now used for new and replacement PWR steam generators.

For evolutionary water cooled reactors, the basis for achieving high performance is established in the design phase. For example, design for short outages, for on-line maintenance, and for an overall goal of simplicity should contribute to high availability. Advances such as better man-machine interfaces using computers and improved information displays, greater plant standardization, and better operator qualification and simulator training, applied at current plants, will contribute to high performance of future plants. Improved availability will be gained by increased design margins that accommodate disturbances and transients without causing reactor trip, and provide additional assurance that plant lifetimes of 60 years can be achieved.

Technological features are being incorporated to meet increasingly stringent safety objectives, by improving accident prevention and mitigation. Many new design features in evolutionary plants have been tested to demonstrate technology readiness.

Target power generation costs, for competitive evolutionary plants to be built in the next decade, have been identified in a 1996–97 study done by the OECD in co-operation with the IAEA using cost estimates for electricity generation with fossil fueled base-load plants expected to be commissioned by 2005–2010. From this study, coal and gas-fired plants have a range of generation costs (busbar costs) between 30 mills/kW·h and 42 mills/kW·h (US mill of 1.7.1996), when the discount rate is 5%. At a 10% discount rate the range becomes 32 mills/kW·h to 52 mills/kW·h. These values are country-specific and the results indicate that the closer the nuclear power generation cost is to the lower end of the range, the more locations there will be in which the nuclear plant will be economically competitive.

To meet this competitiveness challenge, construction delays must be avoided, regulatory procedures and requirements must be stable, design must be substantially completed before the start of construction, and construction and operations management must be highly competent.

The financing of new nuclear plants will be strongly influenced by the competitive economics of new combined cycle gas turbine (CCGT) plants; the deregulation of the electricity markets in many countries; and by the move towards private ownership of utilities. The introduction of high efficiency CCGT plants combined with the low price of gas makes it increasingly difficult to demonstrate that nuclear power plants are competitive. The problem is compounded by the increasing deregulation of the electricity markets with a departure from vertically integrated generation and supply companies which gave an assured revenue stream from the electricity produced.

For nuclear plants, principal preconditions to financing are: national policy supporting nuclear power; creditworthiness; economic competitiveness; project feasibility; assurance of adequate revenues by long-term purchase agreements; and no open-ended liabilities. Special commercial risks must be considered, including the size of the investment, long lead and construction times, complex technology, regulatory uncertainties and political risk. These risks are likely to result in a premium on the financing rates.

For domestic projects, conventional financing is likely to be adopted, with a mix of equity and debt financing. The latter may comprise: bond issues, domestic bank credits, and in the case of state-owned or controlled utilities, credits from public entities or government funds. When importing a nuclear plant, the conventional approach to financing the imported portion is to invite financed bids. Export credits typically form the basis of the foreign financing package, because these generally have the most favorable terms and conditions. Suppliers from several countries may join in a consortium subdividing the supply and involving several export credit agencies. For imported projects, the work performed by domestic companies and labor force is usually financed locally.

The accidents at Three Mile Island and Chernobyl strongly contributed to the public's negative image of nuclear power. Today's international approach to nuclear safety should help assure that mistakes of the past, in part resulting from lack of openness, are never repeated. The activities of WANO, whose membership includes all organizations operating nuclear plants, with its mission of maximizing safety and reliability of nuclear power plant operation by information exchange, comparison, emulation and communication among members; and the IAEA's activities including OSART missions, establishment of internationally recognized safety standards and requirements, and the co-operative agreements within the Convention on Nuclear Safety, are helping to achieve this goal.

The news media has a strong influence on public opinion, but its approach differs from the scientific approach taken within the nuclear community. The media sometimes does not present balanced and objective information on nuclear power issues, so broadening the dialogue with interested groups to disseminate information outside the nuclear community is necessary to give nuclear power a fair hearing. To help regain public acceptance the nuclear community must focus on:

- maintaining a high level of safety at operating plants
- describing nuclear safety to the public in clear terms
- clarifying the health effects of low-level radiation
- further developing technologies for assuring a high degree of safety
- demonstrating and implementing high level waste disposal
- expressing the benefits of concentrating and disposing nuclear waste in contrast with the dilution and dispersion approach of fossil fuel burners.

Advances in technologies related to nuclear safety

Technological advances are incorporated into evolutionary designs to meet increasingly stringent safety objectives. Development has been carried out worldwide on new systems for heat removal in the event of an accident. Progress has been made in containment design and in instrumentation and control systems, and design features are incorporated to improve prevention of severe accidents and to mitigate their consequences.

Current water cooled reactors, and some evolutionary designs rely, in the event of an accident, on redundant and diverse active systems to transfer decay heat from the core and primary system and finally from the containment building. A high degree of reliability and safety with such traditional safety systems can be achieved through redundancy, separation, and diversity, and by assuring with high confidence the supply of electric power for their operation. Other evolutionary designs incorporate safety systems relying on passive means with gravity, natural circulation, and compressed gas providing the driving forces. Heat is transferred to either evaporating water pools or to structures cooled by air convection. Passive systems can simplify safety systems, improve reliability, mitigate the effect of human errors and equipment failures, increase the time operators have available to cope with accident conditions, and reduce reliance on off-site power supplies. They have an advantage in areas that can be contaminated in a accident, since such areas may be inaccessible for repair. However, passive systems have lower driving forces and less operational flexibility.

In some evolutionary concepts, there is a combination of active safety systems and passive safety systems. Some designs use passive systems to back-up active systems. The balance between active and passive systems is plant-specific and must weigh validation for plant conditions, integration into the overall plant safety systems, in-service inspection requirements, maintenance, reliability and the impact on costs. Adequate testing of passive systems is important to determine conditions affecting their performance, to establish their reliability, and to provide data for validation of computer codes used to predict plant response to accidents. This is especially important for the relevant low pressure and low driving forces associated with passive systems.

The containment is a key component of the defense-in-depth philosophy, since it is the last barrier to prevent releases of radioactive material in the event of an accident. For several evolutionary designs, additional margins in containment design are included to accommodate complex loading in case of severe accidents. The containment can be designed in many ways to meet safety and economic goals. The types of containment designs are in the categories of prestressed or reinforced single concrete containments with a steel liner; cylindrical and spherical steel containments; and prestressed double containment with and without a steel liner. The early designs of smaller reactors used steel containments, but for larger reactor designs, the requirements on steel containments have become difficult to satisfy and concrete containments are more common.

In spite of the high level of safety achieved by current plants, safety objectives for future plants include an enhanced level of safety with respect to prevention and mitigation of severe accidents. A recent review of trends in the development of water cooled reactors has resulted in a set of severe accident challenges commonly considered in new plant designs. Among these are challenges from high-pressure melt ejection and direct containment heating, hydrogen combustion, steam explosions, and core-concrete interactions. While the phenomena are complex, there is generally sufficient understanding to design features to cope with the conditions that would result from a severe accident. The advances in understanding important challenges are, in many cases, leading to common approaches being employed by designers. Early containment failure due to high pressure melt ejection is typically prevented by incorporating means to reliably depressurize the primary system prior to vessel melt-through, and direct containment heating is minimized by arrangements to collect and confine the molten core debris. Common strategies are also found in the area of preventing or mitigating hydrogen combustion in PWRs and HWRs; typically, this involves large volume containments, and installation of igniters and/or autocatalytic recombiners to burn hydrogen at low concentrations, reducing the resulting pressure rise in containment. An alternate approach incorporated into BWR designs is to inert the containment atmosphere with nitrogen. This eliminates the potential for fires.

In other areas, such as steam explosions and debris coolability, research has improved the understanding of phenomena on preventing or mitigating the challenge to the containment. For ex-vessel steam explosions, strategies range from maintaining a dry reactor cavity prior to and during melt relocation, designing capabilities for ex-vessel cooling to prevent melt-through, or demonstrating that the cavity design will withstand the potential steam explosion. With respect to coolability and prevention of core-concrete interaction, there are various strategies; one concept has a large spreading area and an overlying water pool that can flood the debris and arrest core-concrete interaction; another provides a core catcher concept with cooling by water from below.

The nuclear industry is taking advantage of the rapid developments in electronics, computers, software, and instrumentation and control technology. These new technologies allow more sophisticated and efficient treatment of measurements and control signals and high flexibility and versatility. Equipment in current plants is being replaced as it becomes obsolete, and for the evolutionary water cooled reactor designs modern instrumentation and control equipment is a fully integrated feature. The new systems offer flexibility, essentially unlimited functions and storage, no drift and the capability for advanced diagnostics and automatic back up. However, the new systems are more complex, leading to increased requirements for validation and verification.

Application of the new developments and technologies is expected to reduce the probability and consequences of all accidents including severe accidents and enable evolutionary water cooled reactors to achieve safety goals set for future decades.

Key developments of evolutionary plant designs

The nuclear plant supply industry is proceeding with design and development of evolutionary reactors, based on continuing dialogue with utilities to incorporate experience feedback from current plants. For evolutionary designs, there is a general drive for simplification, larger margins to limit system challenges, longer grace periods for response to emergency situations, and improvement of the man–machine interface systems. All incorporate design features to meet stringent safety objectives by improving severe accident prevention and mitigation. Several of these designs have reached a high degree of maturity. Nuclear regulatory authorities have certified some designs, and some are entering an optimization phase to reduce capital cost. In some cases design optimization leads to higher plant output to take advantage of the economy of scale, while in other cases, economic competitiveness is pursued through simplification resulting from reliance on passive safety systems.

In Session IV, reactor design organizations involved in development of evolutionary water cooled reactors presented the key developments of their designs as follows (in order of decreasing size):

EPR	Nuclear Power International (Framatome/Siemens)
Advanced PWR	Mitsubishi and Westinghouse
System 80+ PWR	ABB Combustion Engineering
BWR 90	ABB Atom
Advanced BWR	Hitachi, Toshiba and General Electric
Korean Next Generation Reactor	KEPCO and Korean Industry
European Passive Plant (EPP)	Westinghouse/Genesi
SWR 1000	Siemens
CANDU-9	AECL
CANDU-6 (E)	AECL
WWER-640	Atomenergoproject — St. Petersburg/Gidropress
AP-600	Westinghouse
AC-600	Nuclear Power Institute of China
AHWR	Bhabha Atomic Research Centre

Brief remarks on the objectives, key features, and design status for these designs are given below:

EPR: Nuclear Power International (Framatome/Siemens)

The EPR is designed to satisfy the European Utilities Requirements and to meet common safety requirements of the German and French safety authorities. The design effort benefits from the feedback of operating experience from the more than 100 nuclear power plants designed and constructed by Siemens and Framatome. The basic design was completed at the end of 1997. As a result of a design optimization phase carried out in 1998 to achieve competitive economics, the power rating has been increased to 1750 MW(e) to take advantage of economy of scale. The major design features of EPR that are targeted towards the two key objectives of safety enhancement and cost reduction are: simplification of safety systems and elimination of common mode failures; increased grace periods for operator actions by designing components with larger water inventories; and taking measures to further limit the consequences of severe accidents so that relocation or evacuation of the public in the plant vicinity is no longer necessary. A high degree of safety is achieved mainly through incorporation of well proven active safety systems and diverse backup systems. One evolutionary feature for severe accident mitigation designed to prevent molten core-concrete interaction is a corium spreading compartment provided with a protective layer and with provision for active cooling of the basemat and passive flooding of the compartment with water after corium spreading. Measures to ensure economic competitiveness include the higher power rating, design for high plant availability (predicted by NPI to be 92%) and thermal efficiency (predicted to be 36%), reduced construction time, and a design lifetime of 60 years. A significant reduction in fuel cycle cost is foreseen due to a high burn-up (exceeding 60MWd/kg HM) core.

Advanced PWR: Mitsubishi and Westinghouse

In Japan, the APWR was developed as a standardized design under the organization of the Ministry of Trade and Industry, with involvement of Japanese utilities, Mitsubishi Heavy Industries and Westinghouse Electric Corporation, and incorporating experience with construction and operation of current PWRs. Japan Atomic Power Company plans to build two APWRs at the Tsuruga site. To take advantage of economy of scale, the power level was increased from 1420 MW(e) to 1530 MW(e) by increasing the core flow without increasing reactor vessel size or steam generator surface area. Key evolutionary features include an improved radial reflector to increase neutron economy and reduce fluence to the reactor vessel and other internals to help assure the 60-year design lifetime. The APWR adopts a passive advanced accumulator design (pressurized with nitrogen gas) to enhance the ECCS reliability and contribute to simplification (i.e. elimination of the low-head injection pumps of current designs). The advanced accumulators inject cooling water at a high flow rate at the early stage of a large break LOCA (as do conventional accumulators), and then passively adjust the injection flow to a lower rate thus fulfilling the role of (active) low-head injection pumps of current designs. Performance of the advanced accumulators has been confirmed by test.

System 80+ PWR: ABB Combustion Engineering

ABB Combustion Engineering uses evolutionary design improvement and construction processes to address economic and safety objectives, relying on proven components and systems. Improvements are implemented in relatively small steps in actual construction programmes to maintain high confidence that plant start-up and commercial operation will proceed as expected. This approach has been applied for the standard System 80 plant (e.g. Palo Verde), in the on-going construction programme for the Korean Standard Nuclear Plants (KSNP) designs (based on System 80), and for the 1350 MW(e) System 80+ standard plant design. The System 80+ has been designed to meet the EPRI URD. ABB CE received design certification for the System 80+ design from the US Nuclear Regulatory Commission in 1997. Features to increase redundancy, diversity and simplification are included in the System 80+ emergency feedwater system (EFWS), the safety depressurization system (SDS), the safety injection system (SIS), the containment spray system (CSS), the cavity flooding system (CFS) and others. The EFWS consists of two divisions, each with two emergency feedwater pumps, and one storage tank. The SDS rapidly vents steam from the pressurizer to permit feed and bleed cooling of the reactor core after total loss of feedwater, and to reduce primary system pressure after a core melt to prevent high pressure core melt ejection. It consists of two redundant piping trains from the pressurizer to spargers in an in-reactor storage tank (IRWST) with valves powered by diverse electrical power sources. The improved EFWS and SDS are included in the KSNP construction programme as well as in the Korean Next Generation Reactor (KNGR) programme. The SIS includes 4 high-pressure pumps that take suction from the IRWST, and 4 medium-pressure tanks charged with pressurized nitrogen that inject water passively. The need for low-pressure safety injection pumps is eliminated. The CSS consists of two divisions each with a spray pump taking suction from the IRWST. The CFS provides water to the cavity to cool molten corium, taking water from the IRWST via motor operated isolation valves. The improved SIS, CSS and CFS are included in the KNGR design.

The Nuplex 80+™ advanced control complex is incorporated into the KSNPs in the Republic of Korea; three are operating and five are in various stages of design, construction and start-up.

BWR 90 (and BWR 90+): ABB Atom

ABB Atom of Sweden developed the BWR 90 (1374 MW(e)) based on the design, construction and operation of six BWR 75 plants. A smaller version at 1190 MW(e) is also available. While maintaining proven design features, changes were introduced to incorporate technological progress, meet new safety requirements and achieve simplification and cost savings. A new fuel design improved margins to allow higher power operation. The number of welds in the reactor vessel was reduced, leading to less inspection requirements during outages. The design also incorporates advanced I&C systems based on micro-computers, internal recirculation pumps, fine-motion control rod drives and physical separation of the 4-train safety systems, two of which suffice to cope with accident conditions.

The BWR 90 design is available for deployment. Utility requirements were introduced by co-operation with TVO of Finland that operates two of the six BWR 75 plants, and more recently, the design was subjected to a comprehensive review by the EUR.

The BWR 90+ design builds closely on the BWR 90, but with design modifications to incorporate EUR and specific Finnish safety requirements. As an example, severe accident mitigation means were improved significantly, including the use of a "dry" core catcher whose structure is submerged into the containment pool to allow cooling by the surrounding water, containment design to accommodate pressure buildup from hydrogen generation in core-melt accidents, and filtered containment venting for ultimate over-pressure protection of the containment. Other features include increased power level to (1500 MW(e)) and reduced construction time (1500 days).

Advanced BWR: Hitachi, Toshiba and General Electric

Development of the Advanced Boiling Water Reactor (ABWR) started in the 1970s to meet Japanese utilities need for a high performance BWR. GE, Hitachi, Toshiba, the Japanese government and the utilities joined together to develop the ABWR. Beginning in the mid-1970s, many new technologies were tested prior to selection of the design features on the basis of the "test before use" approach. Imported technology, even though it may have had significant operational experience outside of Japan, was subjected to the test series of the ABWR suppliers. Tokyo Electric Power Company adopted the design for units 6 and 7 of the

Kashiwazaki-Kariwa power station; these two 1315 MW(e) units started commercial operation in 1996/1997. New features include a reactor internal pump, fine motion control rod drives, a fully digital control system with operator-friendly main control room, a high efficiency turbine plant with moisture separator reheater, and a reinforced concrete containment vessel for which a steel liner for leakage prevention, with the reinforced concrete providing pressure containment. The ABWR also has a 3-division emergency core cooling safety system to enhance the core injection capability, and its redundancy.

Two more ABWRs are under licensing review in Japan, and several more ABWRs are planned. In the USA, the ABWR has been designed to meet the EPRI URD. Design certification for the ABWR by the US Nuclear Regulatory Commission was issued in 1997, and two units are under construction in Taiwan, China.

Korean Next Generation Reactor: KEPCO and Korean Industry

The Korean Next Generation Reactor (KNGR), a 1300 MW(e) evolutionary PWR, is being developed by KEPCO and the Korean nuclear industry. A number of innovations and improvements allow its power to be about 40% higher, and its core outlet temperature to be lower, compared to the Korean Standard Nuclear Plant (KSNP). Design goals include a 60-year plant life and 90% availability. Improvement include an integrated reactor head assembly that contributes to a reduction of refuelling outage duration and reduction of personnel exposure, use of Inconel 690 in steam generator tubes, improved operability and maintainability of the engineered safety system, increased redundancy of the safety injection system which discharges directly into the downcomer of the reactor vessel, and an in-containment refueling water storage tank that completely surrounds the reactor cavity. These features increase safety system reliability and simplicity, resulting in a core damage frequency estimate that is one order of magnitude lower than that of the KSNP. An optimized design for KNGR is expected to be completed in February 1999, to be followed by a detailed design for standardization. According to the mid-and-long term construction plan of power plants in the Republic of Korea, the first KNGR is scheduled for operation in 2010.

European Passive Plant (EPP): Westinghouse/Genesi

In 1994, a group of European utilities, together with Westinghouse and an Italian consortium including ANSALDO and FIAT, initiated a programme designated European Passive Plant (EPP) to evaluate Westinghouse passive plant technology for application in Europe. Phase 1 of the programme evaluated the AP-600 and SPWR designs against the European Utility Requirements (EUR) and prepared the EP1000 design that conforms to the EUR and is expected to be licensable in Europe. The reference plant design follows the SPWR for the NSSS and containment while the auxiliary systems are based on the AP-600. Phase 2A of the programme, to be completed at the end of 1998, is to improve the design and produce a preliminary cost estimate. Design goals include passive safety systems that need no operator action for more than 24 hours after an accident, a plant lifetime of 60 years, and a predicted overall plant availability greater than 90%. The containment vessel is a freestanding steel cylinder surrounded by a reinforced concrete shield building that provides protection against external events.

The EP 1000 safety philosophy is based on inherent margins (e.g. larger volumes of water, lower power density, negative power and temperature reactivity coefficients) to limit system challenges. Active systems are used as first level of defense against the most probable accidents, with passive systems as the second line of defense. The EPP has a passive injection and core cooling system and a passive containment cooling system. Another design goal is the ability to retain a molten core within the reactor vessel by flooding the reactor cavity, transferring heat from the external surface of the vessel.

SWR 1000: Siemens

The SWR 1000, an advanced BWR, is being developed by Siemens under contract from Germany's electric utilities. The project is currently in the basic design phase to be concluded in mid-1999 with the release of a site-independent safety report and costing analysis. Development goals are competitive costs, use of passive safety systems to further reduce probabilities of severe accidents, assured control of accidents so emergency response actions for evacuation of the local population are not needed, simplification of plant systems based on operating experience, and planning and design based on German codes, standards and specifications set by the Franco-German Reactor Safety Commission as well as IAEA guidelines and the European Utility Requirements. These goals led to a plant concept with a low power density core, with large water inventories stored above the core inside the reactor pressure vessel, in the pressure suppression pool, and in other locations. All accident situations arising from power operation can be controlled by passive safety features without rise in core temperature and with a grace period of more than three days. In addition, postulated core melt is controlled by flooding the reactor cavity and removing heat from the external surface of the reactor vessel, to retain the molten core within the vessel. Specific passive systems are an emergency condenser to remove heat from the core, a condenser to remove heat from the containment, and pressure pulse transmitters to initiate reactor scram, containment isolation of main steam lines and valve operations. The passive systems have been successfully tested using large scale components in test facilities at Paul Scherrer Institute in Switzerland and at the Juelich Research Centre in Germany. It is anticipated that the SWR 1000 can be offered commercially by the year 2000.

CANDU-9 and CANDU-6 (E): AECL

The CANDU 9 is a 935 MWe reactor based on the multi-unit Darlington and Bruce B designs with some additional features from AECL's engineering and research programmes. AECL has submitted the design to the Canadian nuclear regulator (AECB) for review, and it has been confirmed that there are no conceptual barriers to licensing in Canada. Emphasis is given to use of proven systems and components. CANDU 9 also incorporates an advanced control center with better operability. Engineering and construction techniques are similar to those used for the CANDU 6E.

The 700 MW(e) CANDU 6E, AECL's Enhanced CANDU 6, is an evolutionary design that draws on the experience with seven CANDU 6 units in operation. The CANDU-6(E) incorporates an advanced control and man-machine interface system, and includes an integrated series of passive heat sinks - an emergency secondary-side condenser, containment air coolers, and a moderator heat removal system. Heat removal is achieved by use of the passive emergency water supply (PEWS) tank located at a high elevation in the reactor building. The PEWS provides a heat sink for natural circulation cooling of the moderator and for containment atmosphere cooling. In both cases, the PEWS is a passive backup to active systems. The CANDU 6E is optimized for a new fuel bundle design that provides improved thermal margin and the option of using alternate high burnup fuels. CANDU 6E includes system simplifications to reduce the number of components, simplify control, improve equipment access and maintainability, and improve plant reliability. It also incorporates a high efficiency turbine cycle. Developments for the CANDU-6(E) include an improved shield cooling system to enhance its inherent heat sink capability for severe core damage accidents, a high-pressure design containment to eliminate the requirement for fast acting pressure suppression spray, increased redundancy of the emergency feedwater system, and improved pressure tube materials.

CANDU 6 projects have established the application of advanced engineering tools in the design; the complete plant is configured in a series of 3D-CADDS databases. Plant construction is based on open-top construction with extensive use of pre-fabricated skid-mounted modules. CANDU-6 units are under construction at Cernavoda-2, Wolsong-4 and Qinshan 1 and 2.

WWER-640 (V-407): Atomenergoproject - St. Petersburg/Gidropress

Design of the V-407 WWER conforms with the regulatory documents valid in Russia, with regard for the IAEA Safety Guides. As a result, the design uses technologies demonstrated by practice, increases in reliability and number of defense-in-depth barriers, application of passive safety principles and improvement of inherent safety properties. Important aspects of the design are decreased core power density, large water inventories in the reactor vessel and pressurizer, advanced I&C system, design for 50–60 year plant life, and systems for mitigating severe accidents including hydrogen control and the ability to flood the reactor cavity to prevent vessel melt-through in the event of core melt. Passive systems include: the emergency core cooling system which injects water from a hydro-accumulator and by gravity, from an ECCS water tank; the residual heat removal system which transports heat from the secondary side of the steam generator to the environment; and the cavity flooding system. The only active system supplies highly borated water into the reactor core in the event of an anticipated transient without scram. The WWER-640 has a double containment: an inner steel containment vessel, and an outer concrete containment designed to protect against airplane crashes and shock waves. The construction licenses are issued by the Russian regulatory body Gosatomnadzor for the Sosnovy Bor site (near St. Petersburg), and for the Kola NPP-2 site (Murmansk region).

AP-600: Westinghouse

The 600 MW(e) AP-600 was developed by Westinghouse under the US Advanced Light Water Reactor (ALWR) Programme. It is designed to meet EPRI URD, and incorporates a combination of well-proven technologies and new safety systems relying on dependable natural forces. The passive safety systems of the AP-600 are the core cooling system which removes core residual heat, safety injection and depressurization, the containment cooling system, the control room habitability system, and the containment isolation. In-vessel retention of core debris in the event of a severe accident is achieved by flooding the reactor cavity to cool the external surface of the vessel. The AP-600 design simplifies plant systems and operation, inspections, maintenance and quality assurance requirements by greatly reducing the quantities of valves, pumps, piping, HVAC ducting, and other components. A comprehensive test programme was carried out to verify plant components, passive system components, and containment behaviour. AP-600 received its final design approval, which is the last step before design certification, from the NRC in September 1998.

AC-600: Nuclear Power Institute of China

The passive advanced PWR AC-600 is under development in China, based on the design of the 610 MW(e) Qinshan II nuclear power plant. The reactor design is characterized by: use of a low-leakage, low power density core, an 18–24 month fuel cycle, an integral reactor top structure and no penetrations through the bottom portion of the reactor vessel. A number of safety functions are based on or supported by passive features: emergency heat removal is passive; safety injection is passive-active requiring electric power only for low pressure injection and re-circulation; containment cooling is passive, by condensing steam on the inner surface of the steel vessel which is cooled on its outer surface by natural convection of air; and the main control room habitability system is passive. Further improvements include: simplification of process systems; reduction of the number of components; introduction of digitized instrumentation and control systems; and modular construction yielding a reduced construction period. For severe accidents, external cooling of the reactor vessel prevents reactor vessel failure and subsequent relocation of the core debris into the containment.

To achieve self-reliance in nuclear plant design, the Chinese government has funded significant research and development supporting the AC-600 design. Experiments with respect to critical heat flux (CHF) at low flow rates, characteristics of core injection from core makeup tanks, passive containment cooling, passive emergency core heat removal on the secondary side, steam generator-pump integration, and digital I&C systems, as well as several other tests, have been completed.

AHWR: *Bhabha Atomic Research Centre*

The Advanced Heavy Water Reactor (AHWR), under development by Bhabha Atomic Research Centre in India, is a vertical pressure-tube type reactor using thorium and plutonium based fuel. Specific design features of the AHWR with great impact on its viability, safety and economics are: reactor power 750 MW(th); reactor physics tuned for use of thorium-based fuel, with a slightly negative void coefficient of reactivity; 75% of power generated in thorium; plutonium inventory in-core up to a maximum 300 kg; boiling light water coolant in vertical pressure tubes; advanced coolant channel design with easily replaceable pressure tubes; and a fuel burnup target of 20000 MW·d/tonne. To accomplish this, PuO_2-ThO_2 MOX and ThO_2-$U^{233}O_2$ MOX are used in different pins of the same fuel cluster, and the moderator is heterogeneous, consisting of amorphous carbon and heavy water in an 80%–20% volume ratio.

The total heavy water inventory is much reduced compared to the PHWR, and it operates under low pressure. The negative void coefficient simplifies reactor control, and the boiling light water coolant enables steam generator substitution with steam drums of simple construction. AHWR incorporates a number of passive safety features: heat removal in normal operation takes place through natural circulation; ECCS water is injected directly into the fuel channels; containment cooling and isolation are accomplished by passive systems; and a large inventory of borated water in an overhead gravity-driven pool facilitates core decay heat removal, ECCS injection, and containment cooling for three days without invoking active systems or operator action.

Keys to economic viability of evolutionary water cooled reactors

Several factors tend to make nuclear power less competitive than its alternatives. First, natural gas is cheap, and small gas-fired units can be brought on line quickly to meet small incremental increases in load. Second, de-regulation of the electric power industry in several countries is changing the criteria for competitiveness of nuclear power. Until recently, nuclear power's relatively stable and low fuel and operating costs offset the disadvantage of high capital cost due to the ability, in a regulated market, for the owner to recover the investment over several decades through regulated rates. With de-regulation, independent power producers and private investors want to minimize their capital investment and recover their investment quickly. Under these conditions, the capital costs of nuclear plants must be reduced to be competitive. If this challenge can be met, large nuclear power units with stable and low operating and fuelling costs have an opportunity to provide bulk sales of power via extended grid networks.

Several factors over and above an intrinsic low capital cost and a high plant availability contribute to good economics. Replication of a design can reduce capital and operating costs and shorten construction times. Short construction time will result in lower interest charges during construction. Favourable economics depend not only on good plant design and construction practices, but also on good plant management throughout the operating life of the plant.

Lessons from standardized plant design and construction can be clearly identified from the successful experience of the French nuclear programme. The comprehensive French programme was launched by the French Government and EdF in 1973 and the PWR became the preferred reactor type. To limit design and investment costs, it was decided to standardize design features, and technical specifications for components and systems, to attain standard safety reports, fuel management, operating procedures and training. The first series of standardized units consisted of the CP0, CP1, and CP2 series — three versions of three-loop 900 MW(e) units. These were followed by two four-loop 1300 MW(e) versions, the P4 and P'4 series, and finally by the four-loop 1450 MW(e) units of the N4 series. Each plant has some differences to adapt to site conditions, and variations occur in the detailed fuel management.

The experience from the standardization has been very positive; long manufacturing series of components have yielded significant cost reductions, and so have the learning effects of repetitive construction procedures. Simpler training and reduced spare parts stockpiles are typical benefits for the operation. There are potential disadvantages: a problem occurring at one plant may reveal a generic problem for that series, although such cases have been quite few; and standardization can inhibit technological progress.

The benefits of cost reductions through repeated construction of standardized designs will be obtained globally only if safety and utility requirements become harmonized worldwide; then standard designs could be built in many countries with only minor adaptations to local conditions.

Lessons from plant management to help assure economic competitiveness of future plants have resulted from the recent experiences at Ontario Hydro, Canada, which has 20 CANDU units with a total power of 15 020 MW(e). The performance of these plants, installed between the 1970s and early 1990s, has been very good, but worsened in recent years. Their capacity factor was greater than 80% in early 1980s, but reduced to around 60% in 1996 with a consequent reduction in economic competitiveness. Further, the regulator expressed concerns about declining safety margins. In early 1997, an external advisory group was established and conducted an intensive assessment of the nuclear operation. That assessment ranked the operation of Ontario Hydro's plants as "minimally acceptable", noting a number of causes such as: lack of management leadership and accountability; poor safety culture; inadequate training; lack of configuration management; and deficient organization. A recovery programme is now underway, involving the laying-up of eight units to free resources to upgrade the management systems and procedures, and to catch up on the maintenance backlogs, at the remaining twelve. The key lessons that can be drawn from the Ontario Hydro experience include the importance of having the right people, with the right qualifications in the right place, at the right time and the need for all staff to have a questioning attitude and to be committed to a "safety culture". Configuration management is essential - knowledge of the status of the plant must be maintained and all documentation kept up to date; maintenance must be given high priority and be provided adequate resources; defined standards are needed for the conduct of all work; and the directors and senior management must understand the plant and appreciate the consequences of their decisions and actions. Good new designs are not sufficient by themselves to achieve economically competitive and safe nuclear plants - a high standard of operation and maintenance is also necessary throughout the life of the plants.

In Japan, construction and project management for the 1356 MW(e) Kashiwazaki-Kariwa Units 6 and 7 ABWRs has shown that short construction periods can be a reality. For these plants, Tokyo Electric Power Company achieved a construction period of 51.5 months per unit from beginning of concrete pouring until commercial operation. This was achieved through design features, advanced construction methods, detailed engineering at very early stages of the project, and good construction management. Design features with a significant bearing on the construction scheme include the structural integration of the reinforced concrete containment vessel (RCCV) with the reactor building, and reduction of the building volume through optimization of the plant layout. Installation of the traditional steel containment vessel has in the past normally been on the critical path of plant construction, but with RCCV, installation can be performed in parallel with work on the reactor building, yielding a shorter construction period.

Construction was performed with two different methods; at Unit 6 an "all-weather" construction method was adopted, while more emphasis was put on large block construction at Unit 7. The "all-weather" method involved an early erection of the steel structure of the reactor building and covering of the building by a temporary roof to provide good working conditions. In the "large-block" method, components and structures were assembled into large modules while buildings were erected, and when a block was completed the modules were brought in place by means of large cranes. A 3-dimensional CAD system was used to simulate equipment installation to prevent interference during construction. The design for equipment, layout and building structures was detailed at a very early stage, when large-scale modularization was applied.

The fuel cycle can be optimized, within a wide range of criteria, of which economics, sustainability of resources, environmental aspects, and specific national objectives are dominant; no single strategy is optimal for all countries. An important aspect in this area is flexibility, given the historical difficulties in predicting availability and cost of energy resources and fuel cycle technologies, and the uncertainties and variability in many of these factors. Within the short term, the industry is attached to dominant thermal reactor technologies, which have two variants: a cycle closed by reprocessing of spent fuel and subsequent recycling, and a once-through cycle where spent fuel is stored in advance of geological disposal. Over the longer term, the possibilities for optimizing the fuel cycle are more extensive and many of these typically involve fast breeders and introduction of new fuels such as thorium. Given the finite resources of fissile material, care must be taken today to avoid closing the doors for future options.

OPENING ADDRESSES

CONGRATULATORY ADDRESS

Chang-Hee Kang
Minister of Science and Technology,
Republic of Korea

Honorable Dr. Mohamed ElBaradei, Director General of the IAEA, Dr. Hans Blix, former Director General of the IAEA, Dr. Young-Sik, Jang, President of KEPCO and distinguished participants !

I'm very happy to deliver the congratulatory address at this "International Symposium on Evolutionary Water Cooled Reactors", sponsored by the International Atomic Energy Agency and hosted by the Korea Electric Power Corporation.

I would like to express my appreciation to the IAEA and Korean staff members for their dedication in making this event a success.

Distinguished participants!

As all of you are aware, the international community is exerting its best efforts to reduce the emission of greenhouse gases and thereby prevent global warming.

At the Fourth Conference of the Parties to the UN Framework Convention on Climate Change held in Buenos Aires beginning November 2nd, many discussions concerning the prevention of global warming were held.

Although many different alternatives have been presented, nuclear power is regarded as the most economical and energy-efficient option.

Therefore, nuclear energy will gain even more importance, not only as a mass energy source, but also as a means of protecting the earth's environment.

However, I believe, in order for nuclear energy to play a greater role in the future, securing its safety is of utmost importance.

As we witnessed at the Chernobyl reactor in the former Soviet Union and at the TMI reactor in the USA, a big accident can bring about a colossal disaster.

Therefore, in building any new nuclear reactors, significantly enhancing safety should be the highest of priorities.

In this regard, I believe that this gathering which brings together international nuclear experts to discuss such important topics as safety and Evolutionary Water Cooled Reactors is very meaningful.

Distinguished participants !

The Republic of Korea has continuously exerted its efforts in the promotion of nuclear power generation as well as the development of related technology.

As a result, the Republic of Korea has successfully completed the development of the Korean Standard Nuclear Power Plant (KSNP) which has been enhanced in terms of both safety and reliability. The Uljin Unit 3, the first KSNP, successfully launched its commercial operation in August.

Currently, five KSNPs are under construction in the Republic of Korea, while two units are being built in the People's Democratic Republic of Korea under the auspices of KEDO.

Accordingly, KSNPs will be constructed to fill this demand. Also the Republic of Korea is currently carrying out a "National Advanced Technology Development Project" which includes the development of a 1350 MWe Korean Next Generation Reactor (KNGR).

The KNGR is an evolutionary light-water reactor with a capacity greater than that of the KSNP. The KNGR will adopt a seismic design and improved containment in order to significantly enhance safety.

Korean researchers are aiming to successfully launch the commercial operations of the KNGR by the year 2010.

Furthermore, the development of the System Integrated Modular Advanced Reactor (SMART), a 300 MWe small and medium-sized reactor with nuclear desalination applications, is also under way.

I truly hope that this symposium will provide an opportunity for close international cooperation and information exchange regarding the development of evolutionary nuclear reactors.

Finally, I would like to once again express my appreciation to Dr. Mohamed ElBaradei and other distinguished guests for participating in this seminar despite your very busy schedules.

OPENING STATEMENT

M. ElBaradei
Director General,
International Atomic Energy Agency,
Vienna

Minister Kang, Minister of Science and Technology, Chairman Jang, President of KEPCO, distinguished participants: it is a pleasure for me to be in the Republic of Korea and to welcome you to this IAEA Symposium on Evolutionary Water Cooled Reactors: Strategic Issues, Technologies and Economic Viability. I should like to thank the Government of the Republic of Korea and the Korea Electric Power Corporation for hosting this symposium. I would also like to express appreciation to the OECD Nuclear Energy Agency, the Uranium Institute, the Korean Nuclear Society and the Korea Atomic Industrial Forum for their co-operation in organizing this symposium.

The Republic of Korea has been a member of the IAEA since its establishment in 1957 and has served ten times on the Board of Governors. Beginning in the 1960s with active participation in the IAEA Technical Co-operation programme to develop its national nuclear infrastructure, the Republic of Korea has become one of the leading nuclear power countries with fourteen units in operation, six units under construction and a further ten units planned by 2015. This expertise is being shared with other countries through the Republic of Korea's active involvement in various technical co-operation activities including the Regional Co-operative Agreement in Asia.

The Republic of Korea is dependent on imports for over 97% of its energy supply. Nuclear power represents about 27% of total installed capacity and, this year, produced over 40% of The Republic of Korea's total electricity generation. As President Kim Dae-jung stated at the opening ceremony of the Ulchin Nuclear Power Plant Unit 3, the Korean Government has decided that nuclear power will continue to be relied upon in the future as a main energy supply source because of both environmental concerns and the lack of natural resources.

I should add that the Ulchin Unit 3 Plant is the first unit of the newly designed Korean Standard Nuclear Power Plant series to come into operation. It is also the reference reactor for the construction by the Korean Peninsular Energy Development Organization (KEDO) of two light water reactors in the Democratic People's Republic of Korea (DPRK).

I would like to take this opportunity also to acknowledge the efforts of the Korean authorities in the area of enhancement of nuclear safety and the improvement of reliability. This year the Agency is conducting the one hundredth Operational Safety Review Team (OSART) mission. The very first OSART mission in 1983 was to the Republic of Korea to the Kori nuclear plant. Since then, all nuclear power plant sites in the Republic of Korea have been visited by OSART missions. The fact that in 1997 Korean nuclear power plants achieved an average capacity factor of 87.6%, maintaining the remarkable record since 1991 of an average capacity factor over 80%, is clear demonstration that safety and efficiency are two sides of the same coin.

In these opening remarks my intention is to highlight the importance of technology development for the future of nuclear power generation. The subject of this symposium is timely: technological and scientific research and development will have a major impact on the economic, environmental and strategic context in which Governments, energy utilities and civil society at large make their decisions on the future use of nuclear energy. But this requires a two-way process - scientists and technologists should also be aware of the overall context in which these decisions will be made in order to plan and design optimum solutions for the needs of the future.

At the beginning of this year, 437 nuclear reactors operating in 31 countries provided about 17% of global electricity and accounted for the avoidance of about eight per cent of global carbon emissions. Accumulated operating experience for all nuclear power plants reached approximately 8,500 reactor years.

Global energy demand is growing as a result of expanding industrialization, economic development and increases in world population. It is projected to increase two to three fold for developing countries in the next thirty years, depending on the economic growth scenarios. Not surprisingly, the majority of new nuclear plants under construction in 1997 (33 out of 36) were in Asia and Eastern Europe.

Concern about the impact of human activities on possible global climate change is also growing. At the Kyoto Conference last December, industrialized countries agreed to lower their greenhouse gas emissions. Two weeks ago, the Fourth Session of the UN Framework Convention on Climate Change Conference of the Parties in Buenos Aires adopted a two year Plan of Action which establishes deadlines for finalising the outstanding details for implementation of the 1997 Kyoto Protocol.

Energy produced from fossil fuels accounts for about half of man-made greenhouse gas emissions. Except for nuclear or hydro power (which has limited growth potential), there are not yet any other economically viable, minimal-greenhouse gas-emission options for base load power generation. The extensive use of renewable resources for this purpose does not appear to be close at hand. In sum, nuclear power generation is one of the two proven and readily available options for meeting growing energy demand and mitigating greenhouse gas emissions.

Moreover, for many countries, like the Republic of Korea, reliance on nuclear power presents particular advantages for national security of energy supply. The resource base in uranium and thorium, and their possible future utilization, provide an assurance of energy independence for many countries. And the use of nuclear reactors for co-generation applications such as seawater desalination and industrial heat applications, are positive factors that can contribute to meeting national development goals.

The choice of nuclear power and of a particular energy mix are naturally national decisions which take into account the specific situation and requirements of individual countries. But objectively assessed, the environmental, developmental and national security considerations I have just outlined would suggest that nuclear power, together with improved energy efficiency and greater use of renewable energy sources, should continue to be a major component of many national energy strategies. However, the most recent OECD International Energy Agency projections show the nuclear power share of global electricity supply falling to twelve per cent in 2010 and eight per cent in 2020.

This is due to several factors. New nuclear power projects are at a standstill in Western Europe and North America. In many countries, public concern over nuclear safety, particularly waste management, is a critical inhibiting factor on decisions to construct new plants and on the continued operation of existing safe and efficient plants. Several existing reactors are now approaching the end of their original design life. It is not clear how many of these will be extended in service or replaced with new nuclear reactors or other options. And in some countries which are deregulating their energy markets, the high initial capital costs of new plants and the present availability of inexpensive natural gas have tended to focus new investment elsewhere, including on combined cycle gas power plants.

Thus the future for nuclear power presents a very mixed picture. There are compelling reasons why nuclear power should remain an important source of global electricity supply, but in many countries there are also substantial hurdles in terms of public acceptability and economic competitiveness. In the absence of viable alternatives, the world would not be well served if an important energy option were rejected on subjective and emotional grounds. The challenge is to ensure that the nuclear power option is given a full and fair hearing.

Meeting this challenge requires a two track approach. One track is to continue to work on improving the operational safety and efficiency of existing nuclear power plants. This means achieving a demonstrated safety record throughout the entire nuclear fuel cycle, most urgently with respect to the management of radioactive waste and spent fuel. It also means achieving the highest levels of operational efficiency and economic competitiveness while ensuring, particularly in highly competitive deregulated markets, that cutting production costs does not mean compromising on safety.

The second track is to foster research and technological development with the aim of constantly improving every component of current nuclear fuel cycle technology while also developing advanced evolutionary and innovative reactor designs. Some objectives to reach include proliferation resistant fuel cycles to give the highest assurance of no diversion of nuclear material for weapons purposes; new reactor designs with higher efficiency, lower cost and improved safety; lower output reactors for use where large reactors are not attractive and new techniques for on-site and surface storage and for underground disposal of nuclear waste.

Of the 437 nuclear power plants in operation in 1997, 346 were light water reactors totalling 306 GW(e) and 30 were heavy water reactors totaling 16.4 GW(e). Water cooled reactors are a mature and proven technology with an experience base of over 6100 reactor years for light water reactors and over 600 reactor years for heavy water reactors. But to remain competitive in the future and to enjoy public and government confidence, further scientific and technological development of new designs is important.

This is the major purpose of this symposium: to review technological advances and the readiness of evolutionary water cooled reactors to contribute to meeting near and medium term energy needs.

I cannot over-emphasize the importance of your work for the future of nuclear power generation. Global awareness of the need for sustainable energy strategies provides a window of opportunity and a context for the recognition of the role of nuclear power as an environmentally friendly source of energy. But context may not be turned into a commitment, and the window of opportunity may not be seized, without further scientific and technical development to ensure nuclear safety and improve economic competitiveness.

OPENING ADDRESS

Young-Sik Jang
Korea Electric Power Corporation,
Republic of Korea

Honorable Dr. Mohamed ElBaradei, Director General of the IAEA; Dr. Chang-hee Kang, Minister of Science and Technology; Dr. Hans Blix, former Director General of the IAEA; Dr. Stanley Hatcher, President of the American Nuclear Society and Distinguished Participants!

It is a great privilege for me to deliver the opening address on behalf of KEPCO at this "International Symposium on Evolutionary Water Cooled Reactors", which has been co-organized by the International Atomic Energy Agency and the Korea Electric Power Corporation.

This year celebrates the 100th anniversary of the founding of the electrical power industry and the 20th anniversary of the establishment of the nuclear industry. Therefore I believe it is even more meaningful that we are hosting this international symposium this year. I would like to take this opportunity to express my appreciation to the world-renowned leaders and scholars who represent the international nuclear community as well as leaders and experts from Korea for attending this symposium.

This symposium is the second of its kind since the 1993 International Symposium on Advanced Nuclear Power Systems. I am confident that this symposium will provide a valuable forum for reviewing advanced nuclear technologies being developed worldwide as well as discussing measures to enhance the safety and reliability in the development of viable advanced reactors. I also hope that the participants will be able to exchange valuable information on efficient development strategies.

As all of you are well aware, the peaceful use of nuclear energy and the NPT treaty is contributing to the development of civilization and the improvement of welfare in a wide variety of fields including power generation, medicine, agriculture and hi-tech industries.

Nuclear energy is playing a central role in the stable supply of electricity and economic development as an alternative to fossil fuels. Also, nuclear energy contributes to the preservation of the environment through the reduction of greenhouse gases.

Despite this fact, since the TMI and Chernobyl accidents, the majority of western nations are hesitant to carry out new nuclear power programs. Even Asian countries, which have been enthusiastic about nuclear power, are experiencing difficulties arising from the current economic crisis and insecurity of the public.

However, the Republic of Korea, which imports 98% of its natural resources, has been keen on the peaceful development of nuclear energy as a viable alternative to petroleum. Currently, the Republic of Korea has fourteen nuclear reactor units in full operation and eight units currently under construction including two in the North.

KEPCO has maintained an average capacity factor of over 80%. In the case of 1997, the number reached an average high of 87%. This year, nuclear power accounted for around 40% of total power generation.

The development of the 1000MW(e) Uljin Unit 3, the first Korean Standard Nuclear Power Plant (KSNP), was completed in September. The development efforts relied greatly on Korean technology and the cooperation of ABB-CE while the design and construction of the power plant were achieved through the deployment of advanced technologies following stringent safety standards. Both the technology and safety of the KSNP received verification from the IAEA. The 700Mwe Wolsung Unit 3, developed with the cooperation of AECL, is also very safe.

The Republic of Korea has already launched the Next Generation Nuclear Reactor Program, which utilizes the expertise accumulated through past experience.

The Next Generation Nuclear Reactor Program, which was initiated in 1992, has adopted world-class design standards and its reactor type is the evolutionary water cooled reactor. In order to enhance the safety and economic viability, thermal output has been increased to the 4000 MW level while upgrading basic safety systems and prevention and mitigation facilities. Furthermore, the designing stage of the state-of-the-art main control room is at its final stage. Preparations are being made so that the commercial operation of the first Korean evolutionary water cooled reactor can go into commercial operation soon after the year 2010.

KEPCO will strengthen our commitment to the IAEA's peaceful use of energy. Also, we plan to expand technology and information exchanges as well as cooperation with various international nuclear organizations including WANO.

As the Korean economy experienced rapid growth, the electricity consumption rate during the past five years has increased by 11 to 12% annually. Yet after the economic crisis that was triggered by the accumulation of short term foreign currency debt, this growth rate has slowed down by 5%. This has made it difficult to secure investment resources, especially since international lending rates have been prohibitively high at over 20%.

As a result, the Korean government and KEPCO will revise the Long-Term Power Development Plan. However, under the current energy source diversification plan, nuclear power will continue to account for over one third of total power generation. Based on this, total generating capacity of nuclear power must double that of today's by the year 2015. To meet this aim, new units, including the 1,350 MW(e) evolutionary reactors, will be constructed.

In addition, we will spare no efforts to increase the efficiency of existing facilities by enhancing performance. Also, efforts to improve the financial structure and implement a low-cost, high-efficiency management structure will be carried out. We will play a leading role in the revitalization of the economy and the fulfillment of the "Rebuilding Korea Campaign" proposed by the Government of the People alongside the people of the Republic of Korea who have participated in the gold exporting campaign.

KEDO designated KEPCO as the prime contractor for the two 1000 MW(e) light water reactors to be built in the People's Democratic Republic of Korea. Since then about 120 engineers and experts have been sent to Sinpo, the People's Democratic Republic of Korea. Currently, siting is in progress and under the keen interest of President Kim Dae-Jung, construction is expected to begin in the near future.

As you are well aware, this project is aimed at freezing the North's nuclear weapons program. But we have another mission, which is to open a new chapter in South-North cooperation. We are carrying out the Sunshine Policy proposed by President Kim Dae-Jung to ensure that the nuclear community can play a leading role in the Rebuilding Korea Campaign. I sincerely hope that all leaders of the nuclear community who are attending this symposium will continue to have keen interest and extend cooperation in implementing this project and the the People's Democratic Republic of Korean policy of the current administration which proposes "peaceful coexistence and economic exchange with the North."

For the first time in our history, the Republic of Korea is pursuing the hand-in-hand growth of democracy and economic development. However, the Republic of Korea is also facing difficulties in securing additional sites for nuclear power facilities due to the NIMBY (not in my back yard) syndrome and public concern over nuclear safety. Moreover, the extra cost associated with site resident compensation and waste treatment is making nuclear energy over 10% more expensive to generate per KWh than fossil fuels. These issues present a challenge to the further development of nuclear power projects.

Therefore, in order to ensure the continuous development of nuclear energy, safety and economic viability must be enhanced. Transparency of all nuclear projects must be ensured to secure public acceptance and support. Also, in order to garner public support, we must stand by the NPT and never even consider Spent Fuel Reprocessing. I believe this is a common challenge that the nuclear community must meet collectively.

Based on all of the above, "Strategic Issues, Technologies and Economic Viability of Evolutionary Water Cooled Reactors" was chosen as the main theme of this symposium. At the technical session, topics such as "Key Factors in the Decision Making Process", "Advances in Nuclear Safety Technologies" and "Keys to Economic Viability" will be discussed.

I truly hope that all of you will partake in active discussions on new nuclear technologies and that your discussions will contribute to the enhancement of safety and reliability of evolutionary reactors as well as public perception. Also, I hope that this symposium will provide an opportunity to create a brighter future for the nuclear industry and facilitate international cooperation.

And for those of you who have traveled far to attend this symposium, I hope your stay in the Republic of Korea will provide you with an opportunity to see the status of the Korean nuclear industry and experience the Republic of Korea's rich culture and traditions.

Finally I would like to extend my special gratitude to the IAEA and many Korean staff members who have worked so hard to make this symposium a success. Once again, I would like to thank all the participants for being here with us.

KEYNOTE ADDRESSES

(Session I)

Chairperson

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GLOBAL ENERGY DEMAND OUTLOOK

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Abstract

Perhaps the most compelling issue the world will face in the next century is the quality of life of the increasing populations of the poorer regions of the world. Energy is the key to generating wealth and protecting the environment. Today, most of the energy generated comes from fossil fuels and there should be enough for an increase in consumption over the next half century. However, this is likely to be impacted by the Kyoto Protocol on carbon dioxide emissions. Various authoritative studies lead to a global energy demand projection of between 850 to 1070 EJ per year in the mid-21st century, which is nearly three times as much as the world uses today. The studies further indicate that, unless there is a major thrust by governments to create incentives and/or to levy heavy taxes, the use of fossil fuels will continue to increase and there will be a major increase in carbon dioxide emissions globally. Most of the increase will come from the newly industrializing countries which do not have the technology or financial resources to install non-carbon energy sources such as nuclear power, and the new renewable energy technologies. The real issue for the nuclear industry is investment cost. Developing countries, in particular will have difficulty in raising capital for energy projects with a high installed cost and will have difficulties in raising large blocks of capital. A reduction in investment costs of the order of 50% with a short construction schedule is in order if nuclear power is to compete and contribute significantly to energy supply and the reduction of carbon dioxide emissions. Current nuclear power plants and methods are simply not suited to the production of plants that will compete in this situation. Mass production designs are needed to get the benefits of cost reduction. Water cooled reactors are well demonstrated and positioned to achieve the cost reduction necessary but only via some radical thinking on the part of the designers. The reactors of the next century may well be smaller units with higher levels of built-in safety. Whatever the reactors are, the key will be mass production to achieve capital costs of half today's levels.

1. INTRODUCTION

Forecasting the future is a remarkably difficult task - if it were not so, then we would undoubtedly arrange our lives differently to benefit from our knowledge of what is to come. So most forecasts turn out to be inaccurate at best. The task becomes even more challenging in energy predictions because of the unknown magnitude of technological change and the impact of the current thrust towards limiting carbon dioxide emissions.

Perhaps the most compelling issue the world will face in the next century is quality of life in the poorer regions of the world. Dealing with this issue will be made difficult by the fact that it will require enormous economic growth, just to cope with the population growth in these regions, as well as to improve the quality of life. Energy will be an integral part of this complex picture.

Global energy demand will be driven upwards by a number of factors:

1. World population is increasing by 90 million each year, mainly in the developing countries. Demographic experts predict that the population will double by the middle of the next century, to about 10 billion people [1, 2], and that nothing will stop that doubling, short of a disastrous global epidemic affecting hundreds of millions of people.

2. A technological revolution is underway, and is seen clearly in communications. Television already brings images to and from all parts of the world and is an important part of the revolution in communications, particularly in the poorer countries. Improved communications generate justifiable expectations for a better quality of life.

3. Quality of life depends on societal wealth, and wealth is created by the use of resources and energy to generate economic growth. The wealthy countries use large amounts of energy per

Table 1. CONSUMPTION PATTERNS FOR COMMERCIAL ENERGY

		1967	1990	1997
Energy Consumption, MTOE [a)]	World	4046	7851	8509
	OECD	2792	4437	4950
	EMEs	503	1862	2547
Energy Consumption, Ej [b)]	World	172	335	363
	OECD	119	189	211
	EMEs	21	79	109
Carbon Dioxide Emissions, MTC [c)]	World	3329	6049	6447
	OECD	2166	3291	3572
	EMEs	462	1612	2173

a) MTOE = Million Tonnes of Oil Equivalent *Source: BP Review 1998 [5]*

b) EJ = exajoules, 10^{18} joules

c) MTC = carbon dioxide expressed as Million Tonnes of Carbon

capita, and their citizens today are healthier, more educated and live longer than any other people at any time in the history of civilization.

4. And while the environment may not be a high priority in most of the world today, its fundamental importance will eventually dictate the way of life world-wide.

The supply of energy is key to the future welfare of humanity. Energy is the key to overcoming the scourges of hunger, disease and poverty - it is essential to generating wealth and it is essential to protecting the environment in a heavily populated world. Balancing economic development with environmental considerations, particularly the emission of carbon dioxide, will be a major challenge for the next century.

2. CURRENT ENERGY PATTERNS

Today, most of the world's energy comes from the fossil fuels, and within the time-frame of the next half century there should be enough for an increase in consumption. Marchetti has shown that there has been a common pattern for energy sources to grow in market share until new sources displace them [3]. The market shares held by coal and oil are now declining, while those of natural gas and nuclear are rising. NAKIĆENOVIĆ et al (1997) point out that, at the global level, the "time constant" for fundamental energy transitions has been of the order of 50 years [4].

Some characteristics of global energy patterns over the last 30 years are shown in Table 1.

Data for 1990 are included, since that is the reference year for the Kyoto Protocol on carbon dioxide emissions [6]. While still at much lower levels than the countries of the Organization for Economic Co-operation and Development (OECD), the Emerging Market Economies (EMEs) are expanding their energy consumption rapidly, from 12% of the world total in 1967 to 30% in 1997. Because of their stronger dependence upon coal as a principal energy source, their carbon dioxide emissions now amount to over one third of the global total.

All regions of the world increased their carbon dioxide emissions between 1990 and 1997, with the exception of the European Union and the countries of the Former Soviet Union (FSU)and Eastern

Table 2. UNITED NATIONS HUMAN DEVELOPMENT INDEX AND INDUSTRIAL DEVELOPMENT

REGION	HDI	Life Expectancy, Years	Annual Electricity use, kWh/cap	Total Annual Energy Use, GJ/cap	GDP, (PPP$)/cap [a)]
Industrialized countries	0.911	74.2	7542	190	16337
E. Asia excl. China	0.883	71.7	3679	100	9934
Latin America & Caribbean	0.831	69.2	1636	41	5982
World	0.772	63.6	2290	61	5990
E. Europe & Former Soviet Union	0.756	68.1	4170	127	4109
S.E. Asia & Pacific	0.683	64.7	572	18	3852
E. Asia	0.676	69.3	991	32	3359
Arab States	0.636	63.5	1355	50	4454
All developing countries	0.586	62.2	814	24	3068
S. Asia	0.462	61.8	445	12	1724
Sub-Saharan Africa	0.386	50.6	476	11	1407
Least developed countries	0.344	51.2	81	2	1008

a) PPP$ = Purchasing Power Parity, the number of units of a country's currency required to purchase the same representative basket of goods and services that a US$ will buy in the United States.

Europe (EE). The European Union held constant, due to a major replacement of coal consumption by gas and nuclear energy. The FSU and EE, often referred to as "reforming economies" which are in a state of transition, decreased by 36% because of a dramatic slowdown in their economies, resulting in less energy consumption.

3. FUTURE SCENARIOS FOR ENERGY

The pattern for the next ten years, and perhaps the next twenty has already been set, since major energy projects require planning of this scale. This paper focuses on energy over the next fifty years, since it is that time-scale that significant change can be made. It will examine a number of energy scenarios for the middle of the 21st Century.

We can start from the very simple concept that on a per capita basis, people in the poorer countries use about one tenth of the energy used by people in the industrialized countries. There are a number of indicators suggesting that their quality of life could be much better if their energy consumption were to increase to about 100 GJ/capita. That would bring them to about half the average for the industrialized world and one third of the current US and Canadian level. For example, the United Nations Human Development Index [7], based on life expectancy, adult literacy, school enrolment, and real GDP per capita, shows a strong correlation with per capita energy consumption (cf. table 2). This is true for all regions of the world, except in the countries of the Eastern European countries and the Former Soviet Union, which are undergoing a major economic reforming. If a future world of 10 billion people consumed 100 GJ/capita, then the global energy demand would be about 1000 EJ.

A second approach is to examine historical patterns in energy consumption. World energy consumption has increased steadily at an average of 2% per year for almost two centuries. Over the last 25 years the average growth has been 1.8% per year [5]. If that growth were to be continued to 2050, the energy demand in 2050 would rise to about 1076 EJ.

A third approach involves a prediction of future energy patterns from economic development and national energy forecasting. One of the most comprehensive and sophisticated studies of energy scenarios has been done by the World Energy Council (WEC) and the International Institute of

Table 3. SOME PARAMETERS OF THE WEC/IIASA 1995 STUDY TO 2050

CASE	A	B	C
	High Growth	Middle Course	Ecologically Driven
Population, Billions	10.1	10.1	10.1
GWP[a)]	100	75	75
Energy Intensity Improvement Primary Energy/GDP, %/yr	-1.0	-0.7	-1.4
Primary Energy Demand, EJ	1065	850	600
Resource availability Fossil Non-fossil	 high high	 medium medium	 low high
CO_2 emission constraint	no	no	yes
CO_2 emissions, GTC[b)]	9-15	10	5
Environmental taxes	no	no	yes

[a)] GWP = Gross World Product in Trillions of US(1990)$
[b)] GTC = Carbon Dioxide Emissions, expressed in Gigatonnes carbon

Applied Systems Analysis (IIASA) [8] (WEC/IIASA, 1995). It starts from earlier work by WEC, using projections made by its individual member countries for energy supply and demand to the year 2020 [9] (WEC, 1993), and extends this through detailed modelling to 2050 and 2100. Table 3 shows the basic parameters of the study to the year 2050.

The range of global energy demand for 2050 in these scenarios is from 1065 EJ to 600 EJ, and the "middle course", which the authors consider more pragmatic, has a value of 850 EJ. More recently, the International Energy Agency (IEA) concluded that a "Business-As-Usual" forecast for global energy consumption gives a virtually linear increase to 615 EJ in the year 2020 [10] (IEA,1998). If this were continued to 2050, the resulting global energy demand would be 855 EJ. Table 4 shows the results of these studies and projections for the years 2020 and 2050.

Thus, regardless of how one examines the future prospects, and how the energy is distributed, it seems that the global demand by the mid-21st Century will most likely be of the order of 850-1070 EJ, unless constrained by government action, such as taxes or other CO_2 constraints. That is nearly three times as much as the world uses today. However, there does not appear to be any fundamental limitation of resources that would prevent such energy use. The planet can support 10 billion people in a reasonable quality of life, if there is the political will to do so.

Table 4. GLOBAL ENERGY PROJECTIONS, EJ/yr

Year		2020	2050
WEC/IIASA (1995)	A	700	1065
	B	600	850
	C	500	600
IEA "Business-as-Usual"		585	
Historical Growth Rate		585	1075
100 GJ/capita			1000

4. ENVIRONMENTAL CONSIDERATIONS

Concern about changing the global environment has emerged in the last decade as the major factor that now challenges the "business-as-usual" economic development of the world. Formalized in the 1992 United Nations Framework Convention on Climate Change, it has led to commitments by many industrialized countries at the Kyoto Conference in December 1997 to reduce their carbon dioxide emissions by up to 8% from 1990 levels. However, the reality is that most countries have increased their carbon dioxide emissions since 1990, with the notable exception of the former centrally planned economies of Europe, whose economies are undergoing severe decreases as they embark upon major restructuring towards free economies. Germany, France and the UK have successfully reduced emissions, primarily through the expansion of nuclear power generation.

As both the WEC/IIASA (1995) and the IEA (1998) studies show, unless there is a major thrust by governments to create incentives and/or levy heavy taxes, carbon dioxide emissions will continue to grow globally. The problem is even more challenging than appears on the surface, since most of the increase in carbon dioxide emissions will come from the newly industrializing countries, who currently do not have the technology or financial resources to instal non-carbon energy sources such as nuclear power, and the new renewable energy technologies.

Even in many of the industrialized countries, there will be increases in carbon dioxide emissions, unless governments impose incentives and/or penalties to modify the natural market force choice of energy sources. As an example, the U.S. Department of Energy has recently published estimates for the energy consumption expected to be used for electricity production [11]. From these estimates one can infer that, under a competitive electricity market, carbon dioxide emissions from the U.S. electric power industry will be from 35 to 52% higher in 2015 than in 1990. This study predicts that no new nuclear plants will be built, and that there will be a gradual decrease in the use of nuclear power for electricity generation as old plants are retired. It notes that there is a variety of proposals under consideration by State legislatures and by the U.S. Congress to support the continued development and use of renewable energy.

5. SCENARIOS FOR 2050

5.1 Prospects for Achieving the Scenarios

Some current results from WEC/IASA are presented in figures 1 and 2, and a comparison with the WEC/IIASA, 1995 study is shown in Table 5.

Population projections are generally comparable with the majority of expert opinion, and the generally accepted projection for 2050 is 10 ± 0.6 billion. All of the scenarios are derived using regional economic growth rates that appear justifiable in light of experience over the most recent decades. Energy intensities have improved steadily in countries with increasing GDP over recent decades.

However, there are severe conflicts in the prospects for achieving the economic growth and energy growth while reducing carbon dioxide emissions. While there has been much talk since 1992 about reducing carbon dioxide emissions, the reality is that, with the exception of the EU, all regions with economic growth have increased their carbon dioxide emissions and continue to do so. Case C assumes CO_2 constraints and environmental taxes. To date there have been no incentives or penalties put in place to curb emissions.

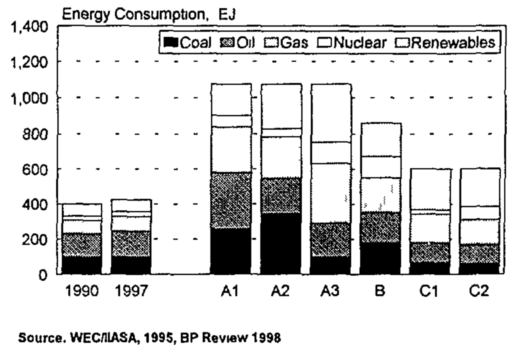

FIG. 1 Global Energy Mix

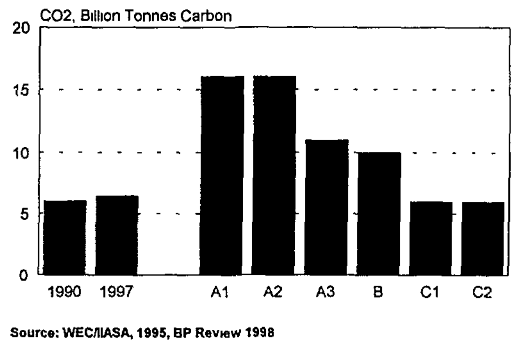

FIG. 2 Carbon Dioxide Emissions

Much emphasis is placed upon the "new renewables" (other than hydroelectric generation or wood burning) to displace some of the fossils fuels, particularly in the U.S. [12]. However, according to the U.S. Energy Information Administration [13], renewable energy in the U.S. contributes 8% of the total energy supply with "conventional" renewables making up the majority of this, and solar and wind amounting to less than 0.2%.

It took over 70 years for oil and gas to reach a market share of 20% in a rapid economic growth period and strong commercial competitiveness (WEC/IIASA 1993). Thus, it seems unlikely that the renewables will grow from 20% (primarily hydroelectric and wood) to 40% in 50 years, particularly when virtually all of this growth is in the "new renewables". Solar and wind technologies are admitted to be not competitive at present (EIA-0623, 1998) [11]. This difficulty in expanding the new renewables is acknowledged in the WEC/IIASA, 1995 study which states that "the two Case C scenarios present challenging global perspectives." The Case B scenario seems more likely to be achievable and the authors note that it is the most pragmatic. Case B calls for an increase in nuclear power of nearly a factor of five relative to today's capacity. However, in this case fossil fuels continue to increase markedly and carbon dioxide emissions are nearly double the 1990 levels.

5.2 Nuclear Power

To achieve the Case B total for nuclear power by 2050 would require an average rate of addition of 40 GWe/yr from 2010 to 2050. From a practical viewpoint there is no reason that this is not achievable. New capacity averaged about 30 GWe/yr for plants ordered in the 1970s and coming on-line in the early 1980s.

One might argue that, while this is a reasonable target, it does not come close to meeting the intent of the Kyoto Protocol. Much larger contributions will be required of nuclear and renewables if carbon dioxide emissions are to be stabilized at levels at all close to 1990 levels.

So, how large a nuclear power supply is practicable? Take, for example, a target for nuclear energy to supply one-third of the global energy demand. For Case B, this would require a nuclear

Table 5. COMPARISON OF WEC/IIASA RESULTS WITH CURRENT VALUES
Global Energy Consumption in EJ

	1990	**1997**	**2050**					
Scenario			A1	A2	A3	B	C1	C2
Primary Energy	400	425	1075	1075	1075	860	600	600
Coal	96	99	258	344	97	181	66	60
Oil	136	146	322	204	194	172	113	107
Gas	76	85	258	237	343	198	162	144
Nuclear	24	27	65	43	118	120	24	72
Renewables	68	68	172	247	323	189	235	217

capacity of about 4000 GWe. Construction of about 100 GWe per year could achieve this over forty years. That's considerably more than the maximum rate of construction achieved to date, but it is feasible from an industrial point of view. Economics and the availability of capital will likely be the critical factors, rather than technology or industrial capability.

There are, of course some obvious prerequisites to such an expansion of nuclear power. First, there must be no major accidents that release radioactivity to the environment on the scale of Chernobyl, within the next decade or two. Such an event would rekindle public fear and might well destroy prospects for nuclear to fulfil its promise of cheap and abundant energy for the world. Safety culture must be paramount, from design, through building and into operations and maintenance.

Public acceptance depends strongly on continuing safe reactor performance, and progress on waste management. One issue in particular must be resolved before this expansion will take place. Electricity producers and the public need to be satisfied that radioactive waste can be handled safely and economically. This is an issue that will require the co-operation and determined efforts of the industry, regulators and the political sector.

As to fuel cycles, it is generally acknowledged that fuel recycle will be necessary to assure adequate nuclear fuel over centuries of utilization. However, its large scale use will also be a question of economics. At present, recycle is not cost-competitive with the use of fresh uranium.

The known reserves of uranium today represent about 40 years consumption at current rates, comparable with about 42 years for oil, and 62 years for gas. Thus, there is no reason to consider uranium resources in any different manner than those of other energy sources. The low price of uranium over the last decade has resulted in very little uranium exploration and virtually no development of processes for recovery from low-grade resources. Even with nuclear power supplying one third of the Case B energy, the cumulative uranium requirements to 2050 are no more than the conventional resources recoverable at less than US$130/kgU [14].

An increase in uranium demand will stimulate exploration and may well lead to major increases in low cost resources, further delaying the advent of large scale recycle. The same argument holds for the fast breeder reactor. So for the moment, fast breeder reactors and fuel recycle are not the issue. When the nuclear industry is running at a very much larger scale, and the price of uranium increases significantly, then the economics may favour these options.

6. THE ECONOMIC IMPERATIVE

While continuing safe operation, demonstrated waste disposal and public acceptance are all necessary for the future of nuclear power, they are by no means sufficient. The key factor is that economics will be the driving force in an expanding global economy. Nuclear power will not serve the world's growing population if the electricity producers will not buy the product.

Utility deregulation is advancing in many of the Western countries. Deficit and debt reduction has become recognized by governments as essential to the future well-being of their citizens. Competition for capital is becoming more intense world-wide. And a rapid return to the bottom line is becoming the key in most investment decisions. All these business realities point inevitably to the need for nuclear plants that can compete economically with the alternatives. For many electricity producers today, short time from ordering to in-service is crucial and the completed capital cost is more important than the long-term cost of the power generated. So to get the electricity producers to buy new nuclear power plants, they must become the economic imperative.

Natural gas will be the economic competition over the next decade or two, as it is today wherever it is available by pipeline. In any country that has access to pipeline natural gas, combined cycle gas turbines are the competitive alternative. They can be built quickly and cheaply and, at today's gas prices, produce by far the cheapest electricity. Some will argue that natural gas prices will increase. However, global gas reserves have increased steadily and by a factor of five over the last three decades, even as annual consumption has tripled. It seems generally agreed in the industry that there will be no shortage of oil and gas at the consumption rates of Case B for a half century.

The real issue for the nuclear industry is capital cost. If nuclear power is to compete and contribute significantly to the reduction of global carbon dioxide emissions, the capital cost must be reduced to the point where nuclear becomes the economic imperative - the first choice for energy producers. A reduction of 50% in capital cost is the right order, combined with a short construction schedule. Such a reduction would put nuclear power into a competitive position with pipeline natural gas and would make nuclear power the economic imperative for electricity producers world-wide.

Certainly none of today's designs will compete with pipeline gas at current prices. So there has to be a radical change in the way we do things. Let's go back to the supposition that the demand for nuclear power would be such that the world would need to build 100 GWe every year for 40 years, starting in 2000. That is equivalent to one new plant coming on line every two or three days! Current designs and methods are simply not suited to production of economic plants at such rates. We need mass production designs that production engineers will take into the factories, where large modules are fabricated that can be assembled quickly on site. We need mass production lines to produce one or two plants per week - a steam generator a day, a pressure vessel per week. It may not be practical on a country by country basis but it can be done on a global basis. We do not need new reactor concepts or new fuel cycles. We need evolution, innovation, change and transition to use existing technology to produce competitive products.

To produce dozens of plants every year, the industry must move from a building mode to a manufacturing mode. We are still at the evolutionary stage that the automotive industry was at the beginning of this century, when all cars were hand-built. The nuclear industry now needs visionaries to do what Henry Ford did for the automobile when he introduced mass production manufacturing. And we need to start on it now if there is to be any hope of nuclear power supplying 30%, or even 10%, of the global energy supply by 2050. It will only be by mass production manufacturing and assembling, rather than building, that we can produce several cost-competitive nuclear plants per week.

The water-cooled reactors are well positioned to achieve the cost reduction necessary. However, it will require some radical thinking on the part of the designers to break away from the current system of engineer, procure, construct to a new paradigm of design, manufacture, assemble. The reactors of the next century may well be smaller units with higher levels of built-in safety. Designs such as 100MWe modular helium cooled reactors, with integral gas turbines are already being explored. Whatever the reactors are, the key will be mass production to achieve capital costs of half today's levels.

Today the industry is in decline in many industrial nations. Yet we know where we want to be in the next century - an industry that can put into operation dozens of new plants each year. The question is "How do we get from here to there?" With the current slump in orders, individual manufacturers find it difficult to afford the investment in new designs for factory manufacture. International co-operation will be necessary to organize the resources to achieve success. The nuclear power reactor industry of the next century will have to be a global enterprise, in line with other large scale industrial activities.

7. CONCLUSION

As a result of population growth and economic growth, global energy demand is likely to increase over the next half century to about 1000 EJ, between two and three times today's level. At the same time there will be mounting concern over controlling the release of carbon dioxide to the atmosphere. Thus there will be increasing interest in non-carbon energy sources, such as nuclear power and renewables.

To compete in the energy market of the future, nuclear power must become the economic imperative for energy producers, world-wide. This will require a major reduction in capital cost to about half today's levels. New designs, perhaps of smaller size, engineered for mass production will be necessary to achieve the required capital cost.

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THE ROLE OF NUCLEAR POWER IN SUSTAINABLE DEVELOPMENT

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Abstract

Today's developing countries, with some three-quarters of the world population, consume only one-fourth of the global energy. In coming decades, the global energy consumption is anticipated to increase substantially, to a large extent driven by the developing world. Responsive long-term energy strategies that exploit energy sources with a minimum of greenhouse-gas emissions need be developed and implemented as rapidly as possible to limit environmental pollution. The energy mix that evolves will not depend only on environmental considerations, but also on economic, technological, supply and political factors. On the global level, fossil fuels will continue to be the major energy source, probably with natural gas as the major component. Nuclear power is currently a significant source of energy supply, but there is no consensus regarding its future role. Its use has stagnated in Europe and in North America, but it maintains its position as a strong option in Asian countries. Economy and security of supply, along with an awareness of environmental benefits, have been principal considerations in the choice of nuclear power and these three factors will also determine its long-term role in a sustainable energy future. Comparative assessments of the full energy chain of energy options consider a number of issues: fuel and land requirements; environmental pollutants; confinement vs. dispersion of waste; greenhouse gas emissions; natural resources; and external costs, e.g. interest and depreciation, waste management, and energy taxes. Such assessments will help clarify the merits of nuclear power.

1. INTRODUCTION

Economic development, social development and environmental protection are interdependent and mutually reinforcing components of sustainable development. Sustainable development requires that the current generation should meet their needs without compromising the ability of future generations to meet their own. In this context policies to combat climate change are viewed as an integral part of sustainable development.

Energy has played and will continue to play a principal role in promoting economic growth and improved human well-being. However, there are also environmental implications associated with energy supply. The challenge is to develop strategies that foster sustainable patterns of energy use, which do not irreversibly degrade the environment.

There is a general understanding that the present pattern of energy supply, based mainly on the use of fossil fuels, is not sustainable with its significant contribution to local and regional environmental degradation and potential climate change. Another factor that needs to be addressed when assessing the sustainability of a continued global dependence on fossil fuels is the limitation of its resource base.

The alternatives to fossil fuels most often proposed by advocates of the sustainability concept are renewable energy sources: solar, wind power and biomass. Renewables are believed to have neither health and environmental impacts, nor any limitation from the point of view of their resource base. However, their low energy density places a limitation on their use for large scale commercial, industrial purposes.

In the debate on sustainable energy future, the role of nuclear power is a contentious issue. Many advocates of sustainable energy supply, who are outside of the nuclear community, do not even consider nuclear, because of public concerns on nuclear safety, radioactive waste and non-proliferation issues. For example, the United Nations Development Program, in its document Energy After Rio does not suggest a specific role for nuclear power except in the most doubtful of terms. On the contrary, most nuclear organisations and related industries see nuclear power as the only mature carbon-free electricity generating option that can be deployed even on a much larger scale than today.

This paper analyses the potential role of nuclear power in the context of the global sustainable energy future. The fundamental features of sustainable energy development are examined in terms of the following compatibility constraints:

- Demand driven compatibility;
- Natural resource compatibility;
- Environmental compatibility;
- Geopolitical compatibility; and
- Economic compatibility.

2. DEMAND DRIVEN COMPATIBILITY

Provision of energy services is essential for economic development and human welfare. The present level of world energy consumption is approximately 350 EJ (1 EJ = 10^{18} joules) per year. The World Energy Council (WEC) projects that world energy demand will increase to a level between 650 to 1200 EJ by the year 2050. This increase is expected to be driven by both by the growth in population and in energy consumption per capita in developing countries.

Today the nuclear power contribution to global energy consumption is around 6%, almost entirely in electricity sector, with 437 nuclear power plants operating in 32 countries. Hydro contributes another 6% to global energy supply, by generating also only electricity. Practically all the rest - more than 87% comes from fossil fuels: coal, oil and natural gas. Fossil fuels play a dominant role in the electricity sector (64 %), and practically have no competing alternatives in other areas of the energy sector, such as, industrial heat generation, transportation, district heating, and so on. Non-hydroelectric renewables - solar, wind, geothermal and biomass - today constitute less than 1 % of the energy supply.

Antinuclear groups argue that the present level of nuclear contribution to global energy is not essential, that nuclear should be phased out and that its share in energy supply may be covered by energy efficiency improvement measures and introduction of renewables. Conversely, nuclear groups believe that nuclear power has the potential to realize a significant increase of its share in the future global energy mix, in particular from the view point of reduction of green-house gas emission in the environment. Is this realistic? Let us examine the issue in a systematic manner.

At present, nuclear power provides some 17% of the world's electricity. However, some individual countries have reached much higher levels of nuclear contribution. For example, France relies more then 75% on nuclear electricity. In fact, several countries in the world use nuclear power to meet over 40% of their electricity needs, including Belgium, Sweden, Switzerland, Ukraine, Bulgaria, Hungary. Thus, if one considers that nuclear becomes the least cost electricity option everywhere in the world and that there are no other constraining factors, its share in the global electricity supply could theoretically approach as high as 75%, as currently in France.

Electricity demand currently constitutes some 33% of global energy consumption and has seen greater growth then overall energy demand. In the coming century the electricity share is likely to increase up to 50%. Demand for more electricity is driven not only by economic development, but also because of the ease of use and cleanliness. In this case, if 60% of global electricity supply is generated by nuclear power, then nuclear would meet 30% of the global energy supply, i.e., five times higher than the present share.

One can also envisage quite a large market potential for non-electrical applications of nuclear energy. Today, only some 0.5% of nuclear power generation is used for non-electric applications. Various district heating, industrial process and desalination applications exist in Canada, China, Japan, Kazakhstan, Slovakia and Russian Federation. The most likely new market for nuclear power is supply of heat for industrial, residential, and commercial uses, in particular sea water desalination.

Fresh water is a vital resource that is becoming increasingly scarce in many parts of the world. Huge urban centers will have to consider very large-scale desalination projects. At this scale of operation, nuclear power will be able to provide fresh water economically and reliably.

Hydrogen is commonly regarded as the ultimate energy carrier because of its high heat value, transportability, and the absence of polluting combustion products. It is uneconomical today, but it may become feasible, with production at first through the use of high-temperature reactors to reform natural gas and later by using nuclear electricity to electrolyze water. Large nuclear stations could produce electricity for general use and hydrogen for transportation.

The above factors justify that with respect to demand driven compatibility nuclear has significant potential for increasing its present relatively small share in global energy supply.

3. NATURAL RESOURCE COMPATIBILITY

The known level of uranium resources (4.3 million tonnes) is sufficient to fuel existing thermal reactors for around 60 years. In addition about 11 million tonnes of speculative (undiscovered) uranium resources would allow to sustain the current level of nuclear generation for another hundred years. However, this time horizon would decrease proportionally with the expansion of nuclear share and global energy demand. So, from the resource compatibility perspective, nuclear power based on the use of thermal reactors does not much differ from oil and natural gas.

It is a well known fact that thermal reactors use less than 1 % of the energy available from natural uranium. Most of the energy remains unused in depleted uranium. Introduction of fast reactors in future may help to overcome this deficiency. They may recycle plutonium accumulated in spent fuel of thermal reactors, and by this to convert the most part of accumulated depleted uranium into the useful energy. With the introduction of fast reactors which utilize uranium resources 60-70 times more efficiently than thermal reactors, the resource base for nuclear power may increase up to the several thousand years, making it sustainable. It is worth noting that, although the technical feasibility of this approach is proven, there are concerns related to associated economics and proliferation issues.

4. ENVIRONMENTAL COMPATIBILITY

Global warming issue: Today the fear of global warming resulting from an increasing concentration of carbon dioxide in the atmosphere is a major global concern. The industrialized countries are responsible for more than two thirds of the current CO_2 emissions. These countries have now agreed in a meeting held at Kyoto in December 1997 to accept binding commitments for reducing their CO_2 emissions, compared to the 1990 level, by at least 5% by the year 2010.

Although at present developing countries have been spared from making any similar commitments, the stabilization of global CO_2 concentration at some acceptable level will not be possible unless they also make an equal effort to constrain the CO_2 emissions. Thus, under increasing international pressure, developing countries are likely to slow down and gradually stabilize their emissions of CO_2, in spite of the need for increased energy services to support their socio-economic development.

Nuclear power generates practically no CO_2. Its current contribution to the world's electricity avoids the emission of more than 0.6 giga tonnes of carbon (or 2,300 million tonnes of CO_2 annually) If the currently existing nuclear reactors were to be replaced by average fossil generation mix, global energy-related carbon emissions released to the atmosphere would increase by some 8 percent instantly.

With the increase of nuclear power share in global energy supply its role in stabilisation of global CO_2 emission would increase, respectively.

Local and regional environmental degradation: Of immediate concern to many countries is the issue of degradation of their local and regional environment. A number of large cities in developing countries are now facing increasing levels of smog caused by NOx and SO2 emissions. And regional acidification due to emissions of sulphur and nitrogen oxides has already become a serious problem in parts of major coal consuming countries, e.g. China and India. This problem is expected to aggravate further and spread to a number of countries in South and East Asia, as the fossil fuel consumption in these densely populated countries increases.

Based on local and regional environmental considerations nuclear power, like renewables, has clear advantages over fossil fuel. It does not release any significant amount of noxious gases or other pollutants. It may be pointed out here that, although some radioactive materials are generated during normal operation of a nuclear power plant and other nuclear fuel cycle facilities, the amounts released in the environment are very small and strictly limited by regulations to levels far below those of any health significance, as laid down by international regulations and guides. How small such releases are may be judged from the fact that, in some cases, even the amount of radioactivity released from a coal-fired power plant exceeds that from the normal operation of an equivalent nuclear plant.

Radioactive waste disposal: There is continuous public concern that nuclear waste cannot be safely managed. However, nuclear waste has distinct advantages as quantities involved are quite small - only some 30-40 tonnes per year for a 1000 MWe nuclear capacity operation, as compared to 3-4 hundred thousand tonnes of ash containing several hundreds of tonnes of toxic heavy metals (e.g., arsenic, cadmium, lead, mercury) produced annually by a 1000 MWe coal fired plant. The small quantities permit a confinement strategy with the radioactive material isolated from environment. In sharp contrast, disposal of the large quantities of fossil fuel waste follows an alternative dispersion strategy. Disposal techniques for nuclear waste exist. Deep underground geological formations, which have not been disturbed for many millions of years, are being considered for disposal of high level radioactive waste in some countries.

It is due to public scepticism or opposition coupled with lack of political will that these solutions have not yet been put into practice in these countries. Until some appropriate policy decisions are taken, the present practice of storing such waste above or below ground will have to continue.

The real problem related to waste disposal may arise in countries without adequate geological formation for waste disposal at their national territory. It may also be very costly to have national repository for countries with small nuclear power programmes. For these countries regional co-operation in spent fuel storage and waste disposal may be of particular importance.

The most convincing demonstration to the public that high level waste can be managed will be the construction and operation of a repository.

Some also have an opinion that increased role for nuclear requires the development of new fuel cycle concepts with drastically decreased amount and toxicity of radioactive wastes

Protecting health: Results of the comparative health assessment of the different electricity generation systems indicate that nuclear power and renewable systems tend to be in the lower spectrum of health risks. A significant health and environmental impact from nuclear power arises only from potential abnormal events.

Nuclear safety: Nuclear power plants are built to high safety standards. Nevertheless, there have been two serious accidents. Many lessons have been learned from the TMI and Chernobyl accidents. They triggered extensive nuclear safety reviews and modernisation of safety systems of all existing reactors. There has been a large ongoing co-operative effort to improve the safety of all operating Soviet designed plants that has included significant modernization of instrumentation and control systems.

There are already more than 8000 reactor years of accumulate operational experience worldwide, equivalent to an average of 20 years of operation for each nuclear power unit. Building on this large experience base, today's reactors incorporate improved safety measures.

Large evolutionary units with power outputs of 1300 MWe and above, which incorporate proven, active engineering systems to accomplish safety functions, and mid-size evolutionary designs which place more emphasis on utilization of passive safety systems are being developed. Designers believe the newest plants would be operable with no more than one severe core damage accident frequency in 100 000 reactor- years of operation.

Over the years a global nuclear safety culture has evolved through international collaborative efforts to strengthen safety regime worldwide. Binding international agreements, codes of practice, non-binding safety standards and guides along with international review and advisory services now exist.

Suffice it to say that, based on the today's experience, an objective comparative assessment of nuclear power and other major options for electricity generation with respect to human health risks associated with the operation of their full energy chains puts nuclear power among the least risky power generation technologies. Still the fear of a nuclear accident and the release of radioactivity weighs heavily on the minds of many. The workable remedy would be prolonged nuclear power operation without any other

5. GEOPOLITICAL COMPATIBILITY

Energy security. Countries which do not have large easily exploitable hydro power potential and are also short of indigenous fossil fuel resources are the most vulnerable group from the point of view of energy security. The oil shocks provided a strong stimulus for the development of nuclear power in 1970s. The risk of an interruption in the supply of natural gas and oil is still not negligible. In the very long run fossil fuel prices are projected to inexorable increase. Since fuel costs account for some 50-80% of generating cost of fossil power plant, the increased fossil fuel prices, will enhance the competitiveness of nuclear power. The consideration of energy security have played a key role in the decision of countries like France, Japan, Republic of Korea and Taiwan China, to go for nuclear power on a large scale. In all likelihood, they will also be major determinants in the decision of other countries placed in similar situation.

By diversifying energy sources, nuclear power can provide a hedge against large increases in the national energy bill. The recent financial crisis in Southeast Asia caused a drastic devaluation of national currencies and corresponding hikes in energy import bills. Countries of the region without domestic energy resources but with a high share nuclear power are less affected by this recent change in the terms of trade.

In the long run accumulated depleted uranium and plutonium in spent nuclear fuel of thermal reactors may regarded as indigenous resources for countries planning development fast reactors.

Non-proliferation issues: The risk of nuclear proliferation is a political problem rather than technical one. For years the IAEA has been operating an effective safeguards system to check any possible misuse of nuclear materials from the safeguarded nuclear facilities under its control, to prevent illicit trafficking of nuclear materials and to ensure strict adherence to various treaties related to international non-proliferation regime. With the approval by the Board of Governors last year of the Model Additional Protocol to the safeguards agreements, the Agency has obtained the legal authority to implement a more effective safeguards system. As of 30 November 1998, 38 countries signed this protocol. These measures – further strengthened - will minimize the risk of nuclear proliferation. However, we should keep in mind, that these nuclear power systems were developed in the 1950s and 1960s, when the concerns about the potential adaptation of these civil systems to support nuclear weapon development were not great.

To facilitate large-scale nuclear power development for sustainable energy it would be essential in supplement to the safeguards arrangements to develop more proliferation resistant nuclear fuel cycles than the present ones. The plutonium accumulation issue is of particular importance if one take into account the need of transition in future to fast reactors and recycling of plutonium. There are at present some conceptual proposals how to build the future nuclear power system base on fast reactors with much better proliferation resistance features. They should very carefully analysed by nuclear community in order to find politically acceptable response to the above issue.

6. ECONOMIC COMPATIBILITY

Economics of electricity is an important factor in the choice of electricity generation technologies. As compared to fossil fuel fired plants, nuclear plants are more expensive to build but much less expensive to operate. The specific capital cost in terms of $/kW installed of a nuclear power plant is typically about two to three times that of an oil or gas fired plant and one and a half times that of a coal fired plant. Because of the much longer construction time for nuclear power plants, interest accrued during construction aggravates its capital cost disadvantage. The operation and maintenance costs of all the plants are comparable but the fuel cost of nuclear plant is much lower - only one quarter to one third as much as that of a fossil fuel fired plant. The net result is that electricity generation costs per kWh from all the four types of plants are currently comparable. The relative economics of electricity generation from nuclear, coal, gas and oil fired plants in different countries may however vary as the plant construction cost, interest rate, discount rate, O&M cost and fuel cost for each type of plant will depend on specific country situation.

Technological progress and changes in environmental protection and safety regulations will have an impact on the competitiveness of nuclear power as compared to fossil-fuelled power generation. Further gains are to be expected in the efficiency of gas fired combined cycle technology, resulting in reduced fuel requirements and hence lower electricity generation costs. Similarly, reactors of advanced designs which are now under development in several countries are expected to have lower construction costs and improved fuel cycle efficiency, leading to an overall decrease of nuclear electricity generation costs.

The implementation of environmental protection measures and policies, including more stringent atmospheric emission limits are likely to increase the costs of fossil power plants that will have to comply with those regulations by adding pollutant abatement technologies and/or relying on higher quality fuels (e.g. low sulfur coal) that are generally more expensive. The cost of nuclear generated electricity will not be affected by such measures.

Financial issues: The high up-front cost of nuclear power plants is a serious deterrent for capital-short countries even in situations where nuclear power has clear economic merit and also looks appealing on energy security and environmental considerations. This has delayed the initiation of nuclear power projects in several countries, prolonged the construction periods of many and, in some cases, even necessitated the abandonment of work on partly completed projects.

With the increasing current trend towards privatisation and deregulation of the electricity sector, the financing of nuclear power plants is becoming even more difficult because of their high up-front costs, long construction times and the higher economic risk associated with investments with long amortisation periods. Private investors are generally more interested in quick financial returns on their investment rather than in long-term economic advantages. And local, regional or global environmental benefits are not an issue of their concern, at least under the current national regulations and policies.

The overall financing environment for nuclear power, even in a deregulated and privatised power sector regime would become more favourable once the new generation of smaller, modularised reactors with reduced capital costs and shortened construction times become available on the market.

The financing of nuclear power in developing countries may also improve with possible introduction in future of the clean development mechanism (CDM) as discussed at COP4 in Buenos Aires in November 1998. The idea underlying this mechanism is to implement GHG mitigation where it is most cost-effective . The investors from industrialised countries selecting a mitigation option outside the area of their national reduction obligation expect to receive emission credits that can be applied against their own emission reduction obligation. In contrast, the developing country partner in a CDM venture, expects both technology transfer and financial assistance.

The above review of economic and financial issues related to nuclear power shows that in the long run the prospects of competing with fossil fuels is good. However, in near term, economic and financial issues may be the main deterrents for nuclear power growth. It is, therefore, crucial for the nuclear industry to demonstrate the competitiveness of existing reactors and come up with new designs that would compete with all other sources on the basis of investment costs in order to attract financing.

Many of the current activities of the nuclear industry including life extension and improving the management of operations, are aimed at increasing the competitiveness of existing reactors.

New developments relating to advanced reactors, discussed at this symposium, will also contribute to increasing the competitiveness of nuclear power in the near future. The overall financing environment for nuclear power, even in a deregulated and privatized power sector regime, would become more favorable once the new generation of smaller, modularized reactors with reduced capital costs and shortened construction times become available on the market.

7. CONCLUSION

All three energy options discussed here: fossil fuels, nuclear power and renewables have their advantages and disadvantages with regard to sustainability criteria for energy systems.

Fossil fuels have a big advantage over nuclear and renewables in respect of demand compatibility. Fossil fuels are used in all countries for provision of all types of energy services at competitive costs. The main drawback of this option is CO_2 emission and limitation of resources.

Renewables have clear advantages over fossil fuels from health & environmental compatibility criteria. Although proponents of the sustainability concept believe that renewables would dominate the energy supply in the long run, it is too early to extrapolate from their current share of less than 1% to a significant share in the future energy mix, due to its low energy density.

Nuclear power based on the use of present generation reactors has clear advantages over fossil fuel in respect of health & environmental compatibility criteria. Nuclear does not pollute the air and does not contribute to CO_2 emissions. But there are drawbacks that could impede an active role for nuclear in a sustainable future energy mix. These include: relatively higher investment costs; public concerns on safety, waste and non proliferation issues; inefficient use of limited uranium resources; and limiting its use to base load electricity generation.

In conclusion, nuclear power has the potential to become an active player in a sustainable energy mix. The challenge to the nuclear community is to come up with a new generation of reactors and fuel cycle technologies for the 21st century. They should be characterised with robust economic advantages including lower investment costs; significantly improved safety parameters; more efficient use of uranium resource and with improved proliferation resistance features.

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NUCLEAR POWER IN ASIA: EXPERIENCE AND PLANS

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Abstract

Asian countries have developed ambitious energy supply programs to expand their energy supply systems to meet the growing needs of their rapidly expanding economies. Most of their new electrical generation needs will be met by coal, oil and gas. However, the consideration of growing energy demand, energy security, environmental conservation, and technology enhancement is inducing more Asian countries toward the pursuit of nuclear power development. At present, nuclear power provides about 30 % of electricity in Japan, and about 40 % of electricity in Korea. These and other Asian countries are presumed to significantly increase their nuclear power generation capacities in coming years. Korea's nuclear power generation facilities are projected to grow from 12 gigawatt in 1998 to 16.7 gigawatt by 2004. On the other hand, China and India have now installed nuclear capacities of about 2 gigawatt, respectively, which will increase by a factor of two or more by 2004. The installed nuclear capacity in the Asian region totalled 67 gigawatt as of the end of 1997, representing about sixteen percent of the world capacity of 369 gigawatt. Looking to the year 2010, it is anticipated that most of the world's increase in nuclear capacity will come from Asia. It is further forecasted that Asian nations will continue to expand their nuclear capacity as they move into the 21st century. For example, China plans to develop additional 18 gigawatt of nuclear power plants by the year 2010. Nuclear power is also of particular interest to a number of emerging Asian countries in view of environmental conservation and mitigation of greenhouse gas emissions in particular. Nuclear power appeals to some countries because of its high technology content. The strength in an advanced technology, such as the technological capability related to nuclear power, contributes to the overall development of the corresponding country's engineering base, enhancement of industrial infrastructure and expansion of well-trained technical manpower pool. However, the rapid development of nuclear power in Asia will be faced with challenges both in the respective countries and the rest of the world. This paper discusses the nature of these challenges and presents recommendations as to how such challenges might be overcome. These include reactor safety, economics of nuclear power, handling of nuclear waste including spent fuel, and nuclear non-proliferation regime as well as public acceptance. Japan, China, India, Pakistan, South and North Koreas as well as other rapidly emerging Asia-Pacific countries will continue to stress the important role of nuclear power in their energy plans. To varying degrees, these plans have emphasized the need of increasing the level of "self-sufficiency" in their energy supply systems. "Self-sufficiency" entails such issues as indigenous enrichment capabilities, reprocessing spent fuel at the back-end of the fuel cycle to recycle plutonium and uranium, and in-country capabilities for nuclear power plant design, manufacture, construction and operation.

1. INTRODUCTION

Man is born to live, to live better and more elegantly by all means. To this end, it is imperative for us to keep our candles lit, to keep the complex wheels of this contemporary civilization well-greased and in continual motion, and also to augment our standards of living by achieving sustainable development.

On the other hand, however, meeting of such material demands must not be allowed to strain the eco-systems surrounding our nest, our vicinity and this world as a whole. In other words, we must do all we can to keep our skies and seas blue, to keep our mountains and fields green, to maintain our waters clear and potable, and to restore the pollution-stricken air to the status quo ante. In this context, we ought to bear in mind what Mahatma Gandhi once said: "Nature can fulfil all of our needs. However, it does not have the capacity to fulfil our desires." For our desires are limitless in nature, and sky is the only limit.

We are entering an era in human history where the energy problem for maintaining our socio-economic systems may not be solved without the use of nuclear energy, and also where the energy problem may not be solved with nuclear energy alone either. Truth be told, nuclear energy will not provide a panacea, a solve-it-all solution to all of the world's energy and other needs, but it will certainly remain an important and an indispensable part of the overall equation.

The pejorative image of the looming nuclear mushroom, long entertained in the terrified minds of people who have lived through the Cold War, is gradually giving way to a more positive view of nuclear power as a source of clean and competitive energy, as we redirect our recent preoccupation with the nuclear sword towards nuclear plowshares in the form of nuclear fuel in a power reactor. To make a reality of this reorientation, human reason must prevail over human avarice, and human wisdom must reign over our less benign impulses, which often warp our lives and history.

In no case must we devolve a polluted environment to our offspring: For our generation to soil the atmosphere and aquatic realm would be indeed criminal, with tremendous consequences down the generations to come. It would be awfully difficult, extremely expensive and also time-consuming to reverse the far-reaching effects of eco-contamination, and its consequences for mankind down the generations would be dire indeed. In this context, it may be right what was stated by Dr. In Soon Chang of Korea Nuclear Environment Technology Institute regarding the relation between nuclear technology and nature, which reads: Successful implementation of nuclear power projects through technical enhancement is our way of expressing love towards nature.

It is our conviction that nuclear is new and clear, that nuclear power is, in short, new-clear energy, especially in the Asian region where rapid economic development coupled with very large, energy-hungry populations, will present an impending and critical issue in the arena of global energy market and climate change perspectives.

Asian economics are now at the stage of transforming their industrial infrastructure from the labor-intensive light industries to the energy- and capital-intensive heavy industries. As a result, the average growth rate of GNP for Asian countries in the past decade has consistently surpassed that of the advanced countries, and this increase in GNP has naturally resulted in high energy demand and much faster growth in electric power consumption.

Along with these development characteristics, Asian energy resources are scarce and the cross-border movement of the energy resources is rather limited because of geopolitical barriers. Some countries, therefore, have very high dependence on foreign energy resources. For example, Korea relies on imported energy for more than 97 percent of her energy requirements. The lessons learned from the two oil shocks in the 1970s and the Persian Gulf War have compelled us to adopt national policies to diversify energy resources. Energy and electricity have become pivotal items in government policy especially with respect to long-term planning and technology export drive.

A common characteristic in drawing up electric power development program in the Asian region, excluding Japan, is to meet the immediate demand by fossil energy resources owing to low initial investment, short construction period and lack of technical know-how. To secure stable supply of electric power in the long run, however, nuclear power has long been considered as an optimal choice.

This paper deals with the future aspects of nuclear energy taking into account a number of considerations such as getting rid of oil's grip, contributing to environmental conservation, reducing greenhouse gas emissions, and benefiting from the spin-off effect of high technology. An attempt is also made to address a number of issues associated with the appropriate role of nuclear power in meeting Asian energy needs in the twenty-first century.

2. ENERGY DEMAND IN THE ASIAN REGION

Asian demand for energy is forecast to double between 1993 and 2010, raising the region's share of global energy consumption from under one-quarter to over thirty percent (see Table I). Such an increase would be driven by a combination of rising population and skyrocketing economic growth. The growth of energy consumption will be especially pronounced for the electricity sector. Over the next 15 years electricity generation in the Asian region is projected to increase by 130

Table I. Asian and Global Energy Consumption (M TOE)

Region	1971	1993	2010
1. Asia, of which:	679	1,874	3,696
China	236	731	1,460
Japan	270	461	651
East Asia	101	431	927
South Asia	72	251	658
2.Other regions	4,319	6,206	8,097
Total (1+2)	4,998	8,080	11,793

Source: World Energy Outlook, 1996 Edition, OECD/IEA, Paris 1996, and data for Japan are from Atlantic Council's estimates.

percent (see Table II). Even so, electricity use per person will remain very low for most of the Asian population.

It is important to place nuclear power in its context as one of the most important and reliable power-generating wherewithals in this region in consideration of its impending needs. The generation mix varies considerably among countries depending on domestic energy resources. Countries such as Japan and Korea, inherently lacking domestic energy resources, cannot but rely more heavily on nuclear power, which currently accounts for between 33 and 40 % of their electricity supplies.

Asian nuclear power increase from 1997 to 2010 is projected to double. Many countries in Asia are presumed to significantly expand their nuclear power generation capacities in the upcoming years (see Table III). For example, Korea's nuclear power is projected to grow from 12 gigawatt in 1998 to more than 16 gigawatt by 2004. On the other hand, China and India have now installed nuclear capacities of approximately 2 gigawatt each, which will increase by a factor of two or more by 2004.

The installed nuclear capacity in the Asian region totalled 67 gigawatt or 87 units as of the end of 1997, representing about sixteen percent of world capacity. Looking to the year 2010, it is anticipated that the Asian countries will take up a lion's share of new nuclear power generation capacity in the world. For example, China plans to develop additional 18 gigawatt of nuclear power plants by the year 2010.

Table II. Asian and Global Electricity Output (Terawatt-hours)

Region	1971	1993	2010
1. Asia, of which:	677	2,727	6,393
China	138	839	2,210
Japan	385	905	1,523
East Asia	78	564	1,419
South Asia	76	419	1,241
2.Other regions	4,604	9,771	14,514
Total (1+2)	5,281	12,498	20,907

Source: World Energy Outlook, 1996 Edition, OECD/IEA, Paris 1996, and data for Japan are from Atlantic Council's estimates.

Table III. An Overview of Nuclear Power in Asia

Country	Nuclear share out of the total electricity in 1996 (%)	Installed units and capacity in 1996 (%)	Forthcoming units and capacity (net GWe)	Capacity projected for 2004 (GWe)	Capacity projected for 2010 (GWe)
Japan	36.1	54 (46.5)	4 (4.0)	52.3	70.5
Korea	36.1	12 (9.6)	8 (7.0)	16.7	24.7
China	1.2	3 (2.3)	8 (6.57)	2.0	20.0
India	1.9	10 (1.8)	6 (1.75)	4.0	7.6
Pakistan	0.9	1 (0.1)	1 (0.3)	0.4	1.9
North Korea	0	0	2 (1.9)	0.3	2.0
Other	28.8	6 (5.1)	2 (2.6)	7.1	7.8
Total		86 (65.4)	32 (24.1)	82.8	134.5

Source: Journal of Nuclear Engineering International (1997). Capacities for 2010 in Japan, China and Korea are from Energy Supply Outlook by MITI of Japan (1996), China's Nuclear Energy Outlook by NEI (1997), and KEPCO's internal energy plan (1998), respectively, and other countries' data are from Journal of Nuclear Engineering International (1997). Forthcoming units and capacity are quoted from American Nuclear Society's Nuclear News, March 1998. Korea Atomic Ind. Forum, "The Development and Management Status of Nuclear Power Plants in the World," December 1997.

Over the longer term, this trend is expected to continue, with Asia accounting for an increasing share of global nuclear generation. It is often remarked that Asia will become the new center of gravity in terms of the nuclear industry, wherein additions to capacity over the next 20 years are projected to account for over three-quarters of the global total.

3. THE STATUS OF NUCLEAR POWER IN ASIA

For the time being, the growth of nuclear power in Asia will be largely confined to those economies with currently operating facilities (see Table IV). Nuclear energy has already provided a substantial share of electricity supplies for Japan and Korea. China started to take a leaping step toward massive power generation scheme by opening its domestic market, and is expected to be the central arena for the future nuclear power development program in the world, although the current nuclear power generation share of the total is trivial.

In India, nuclear power provided under two percent of total electricity supplies in 1997; further expansion is planned. Pakistan was one of the first Asian nations to have a nuclear power plant and plans to commission a second plant in 1998. Other countries (Indonesia, Thailand) in the region have indicated that their energy systems will include nuclear power facilities sometime after 2010.

Japan, which produces 18 % of the global GDP, is equipped with the largest nuclear capacity in the region. Currently 54 reactors or the installed capacity of 45.5 gigawatt supply about one-third of Japanese electricity. In 1997, Japanese power reactors achieved a record-breaking average capacity factor (81.3 %) in her nuclear history. Especially, the average annual capacity factor of 23 PWRs (19.36 GW) was recorded at 83.4 % which is much higher than that of the world average. The biggest obstacle to nuclear power construction is siting, which is rendered complicated by local Nimby phenomena regarding public acceptance of nuclear power.

Japan has developed an excellent legal framework for siting and operating a power plant, which is entitled "Three Power Source-Related Acts". The philosophy of these three acts is based on the enforced implementation of reciprocal flow of electricity and corresponding compensations between the terminals. That is to say that the generated power is transmitted to the recipients in relatively affluent areas by means of transmission lines, whereas compensations, in a reverse flow, accrue to the local residents adjacent to the power station through the intermediary arrangement of law.

Table IV. Nuclear Power Program in Asia
(as of December 31, 1997)

	In Operation		Under Construction		Reasonably Firmly Planned		Total	
	GWe	Units	GWe	Units	GWe	Units	GWe	Units
Japan	45.5	54	0.8	1	6.0	5	52.3	60
Korea	12	14	5.7	6	11.2	10	28.9	30
China	2.3	3	3.3	4	5.4	6	11.0	13
India	1.8	10	0.9	4	2.9	8	5.6	22
Pakistan	0.1	1	0.3	1	1.5	3	1.9	5
Indonesia	0	0	0	0	1.8	2	1.8	2
N. Korea	0	0	2.0	2	0	0	2.0	2
Other	5.1	6	2.7	2	0	0	7.8	8
Total	66.8	88	15.7	20	28.8	34	111.3	142

Source: Korea Atomic Ind. Forum, "The Development and Management Status of Nuclear Power Plants in the World," December 1997. Data for Pakistan are from a local source. Data for North Korea are from the author.

Japan is almost entirely self-sufficient in reactor design, manufacture, construction and maintenance. Japanese and American manufacturers are working together to develop light water reactors, including an advanced pressurized water reactor (APWR) and an advanced boiling water reactor (ABWR). A 1300 MW ABWR design was built by the Tokyo Electric Power Company at its Kashiwazaki-Kariwa site. The first unit of this twin unit plant started commercial operation on November 7, 1996 and the second unit on July 2, 1997.

Japan has a well developed nuclear fuel cycle encompassing fuel enrichment and fabrication facilities at the front-end, and a network of facilities at the back-end of the cycle, including reprocessing and recycling. However, Japan must rely on uranium imports. Longer term plans include recycling plutonium in light water reactors and development of advanced fast breeder reactors, but these plans have been set back by accidents that occurred at the Tokai reprocessing plant and at the Monju fast breeder demonstration plant.

At present, there are three power reactors in operation in China (the domestically designed Qinshan Phase 1 and two French PWRs at Daya Bay). These provide two gigawatt of nuclear power, accounting for one percent of total electricity supplies. However, there are plans to develop a major nuclear power industry with the installation of 20 GW nuclear generation facilities by 2010. The implementation of an ambitious nuclear power development program is underway, which includes two more domestically designed reactors, two imported heavy water reactors, and four more imported PWRs. China National Nuclear Corporation, which operates both its military and civilian nuclear fuel cycle facilities, plans to offer fuel cycle services as well as reactors to the international market. Included in the fuel cycle services would be the output of its gaseous diffusion plant and its new centrifuge plant.

At present, Korea has twelve reactors (of which ten units are pressurized water reactors and two are heavy water reactors) providing 40 percent of the country's electricity supply. Six reactors are under construction and ten more are planned to be operable by 2010 maintaining 40 percent of electricity supply. Virtually all power production is in the hands of the Korean Electric Power Corporation (KEPCO), originally a government-owned utility which is being partially privatized.

A Korean Advanced Standard Reactor has been developed as a model for the next generation reactor which adopts some features of the U.S. and European advanced light water reactor design. Korea has no uranium resources, and has neither enrichment nor reprocessing facilities. Korea relies on Europe, Russia and the United States for its enriched uranium although domestically manufacturing its own PWR and CANDU fuels.

In North Korea, the construction of two 1000 MW-class light water reactors started under the aegis of the Korean Peninsula Energy Development Organization (KEDO), as part of an agreement, under which North Korea suspended the operation of its graphite reactor and halted the construction of two larger units of the same reactor type as well as its chemical reprocessing facilities.

In India, there are currently ten operating nuclear units (two boiling water and eight heavy water reactors). In 1997, nuclear power accounted for about 2 percent of total electricity supplies. All units are in the range of 200 MW capacity. Six more heavy water reactors are in various stage of construction, including two 500 MW plants. Long-term goals include the use of PHWRs to produce plutonium for commercial fast breeder reactors, and developing technologies to use domestic thorium reserves. India has steadily pursued a nuclear self-sufficiency policy.

Pakistan has one PHWR in operation and a PWR nearing completion. Pakistan achieved self-sufficiency in the front-end of the nuclear fuel cycle with its own uranium production, enrichment, fuel fabrication and heavy water production facilities, but has not carried out commercial reprocessing.

4. THE REASONS FOR ASIAN NUCLEAR POWER EXPANSION

Asian nuclear capacity expansion endeavors are based on a number of reasons such as meeting the rapidly growing demand for electricity, energy security, reducing greenhouse gas emissions, and benefiting from the spin-off effect of high technology. Many of the reasons are shared by other countries, but, reflecting the diversity of the Asian region, the weight of the different motives varies from country to country.

The rate of dependence on foreign energy resources out of the total energy supply is extremely high in Far Eastern countries compared to other OECD member countries as shown in Table V. That in Japan and Korea is 85 and 97.5% in 1997, while that in UK, USA, France and Germany is 3, 18, 54 and 55%, respectively.

In the Asia-Pacific region, Indonesia, Brunei and Malaysia are the net energy exporters. Indonesia seems to become an oil-importing country from the early years of the 21st century. In fact, Indonesia hit the highest oil production with the annual production of 83 million tons in 1977, but its production rate is decreasing year after year. For instance, her oil production in 1990 was merely 85% of that in 1977. On the contrary, its oil consumption has increased tremendously having recorded 65% increase over the past 15 years.

The world energy market will be largely dependent upon the energy consumption patterns in China which began importing oil as of October 1993 from exporting status. Along with the two-digit growth rate of economy, the increase of China's energy consumption has shown to be skyrocketing. Should the per-capita energy consumption in China reach that of an average (South) Korean, her total

Table V. The Rate of Dependence on Foreign Energy Resources of the Total Energy Supply

Country	%	Remarks
U.K.	3	North Sea oil
U.S.A.	18	Abundant domestic reserves
France	54	Nuclear energy
Germany	55	Domestic coal
Japan	85	Mostly from the Middle East
Korea	97.5	Mostly from the Middle East

energy consumption rate would surpass that of the United States making the global energy picture rather gloomy. And such situation would transform the Far Eastern waters to be a hot conflicting spot.

In addition to the politically unstable situation of oil-producing Middle East, the sea lane between the oil-exporting harbors in the Middle East and the Asian ports is too long and too vulnerable to possible regionable conflicts.

In the year 2010, the Far Eastern region is expected to import 15 million barrels of petroleum from the Middle East a day, which will represent 20% of the total world oil consumption. For this, several tens of oil tanker fleet may have to pass through the vulnerable sea lane every week adjacent to the heavily armed naval and air bases of India, Malaysia, Indonesia and Singapore. In particular, the Malacca Straight, which is 10 km wide and 160 km long and also is shallow territorial sea, is worth receiving special attention in view of strategic considerations. Most of, if not all, the Asia-Pacific countries rely on Middle East oil which has to be transported through this Malacca Straight.

Furthermore, the same oil tankers have to get through another critical zone, that is, South China Sea, wherein many islands and reefs, such as the Paracel Islands, the Spratly Islands and Senkaku Islands are jointly claimed by many neighboring countries. Therefore, this sea lane is another bottleneck in terms of marine transportation security.

The total energy demand in Korea in 2010 will amount to some 250 million TOE (tons oil equivalent). Even if allout efforts be made for the use of nuclear and LNG, at least 125 million TOE of petroleum may have to be imported annually, and this amount will be 50% of Japan's oil import or 200% of those of Hong Kong. In short, therefore, Korea has to be made free from the oil's grip in order not to be a victim of war or conflict in the sensitive area. It is our conviction that nuclear power can be harnessed to play the mitigating role in this regard.

The energy import bill in Korea amounted to 27.1 billion dollars in the year 1997, and this is an unbearable burden to us. In this context, nuclear power can lessen our heavy yoke of balance of trade payments.

4.1 Energy Security

Several Asian countries, including Japan, Korea and Singapore, have few domestic energy resources and are obliged to depend on imports for most of their energy needs, at a much higher rate than other OECD countries in North America and Europe. China has become a net importer of oil and gas since October 1993. The Philippines and Thailand are also heavily dependent on imported energy. A large part of this imported energy is shipped from the Middle East through long and potentially vulnerable sea lanes, as mentioned above.

Nuclear power, especially if the technology and civil works can be provided domestically, is seen to upgrade the degree of energy security through diversifying energy supplies and lessening the import dependence share. Even though most Asian countries are obliged rely on imported uranium and overseas enrichment services, nuclear fuel can be more easily stored than fossil fuel and can thus contribute to overall diversification of energy supplies.

Even for the economics in the region that are better endowed with energy resources, long-term plans to develop or expand nuclear generation have been drawn up to support their industrialization progress. For example, China and India which have abundant energy resources, particularly coal and hydro, fall under this category. In the case of China in particular, these are located far from load centers along the industrialized sea coast. Nuclear power, which is not dependent on the constant flow of fuel supply over a long distance eases these logistical problems.

Despite being endowed with energy resources, India's oil imports account for one-quarter of total export earnings, thus making the balance of payments to be in a bad shape. Even Indonesia, one of the most richly blessed in energy resources of the Asian economies, has expressed an interest in nuclear power so as to allow its oil and gas for export.

4.2 Environmental Conservation

Nuclear power is also of interest to a number of countries owing to its environmental conservation aspects. Urban air pollution is very high in many Asian cities, being much higher than that in most cities of North America and Europe, and typically exceeds the World Health Organization (WHO) guidelines by a large margin.

Heavy dependence on coal in India and China, has led to particularly acute urban air quality problems. The continued and accelerated use of coal in rapidly growing metropolitan areas in Asia would cause further serious deterioration of urban air quality, and expand the area of already extensive acid rain deposition.

With regard to greenhouse gas emissions, most of carbon dioxide discharge into the atmosphere results from the combustion of fossil fuels. As a viable non-carbon emission alternative, nuclear power is expected to be discussed in the context of global measures to limit the emission of greenhouse gases into the environment.

4.3 Nuclear Technology Enhancement

Nuclear power appeals to some countries in view of its high technology content. Such countries would like not to be left behind in technology that could propel modernization and economic development. These countries also argue that strength in an advanced technology, such as the technical caliber in managing nuclear power, contributes to the overall development of that country's technological base, industrial infrastructure and reservoir expansion of trained manpower.

By the end of 1997 over 8500 reactor-years of operation plant experience had been accumulated by the current nuclear reactor systems of the world. On the basis of this experience, the development of advanced reactor designs is taking place nowadays, which comprise three basic types:

- Water-cooled reactors, using water or heavy water as coolant/moderator,
- Fast reactors, using liquid metal, e.g. sodium, as coolant; and
- Gas-cooled reactors, using gas, e.g. helium, as coolant and graphite as moderator.

About 90 % of the nuclear power reactors now in operation are water reactors such as pressurized water reactor (PWR), boiling water reactors (BWR) and heavy water reactors (HWR). While the designs of advanced LWRs (ALWRs) resemble those of their predecessors, they incorporate new passive safety systems as well as plant simplification. The first and second BWRs of advanced design were successfully connected to the Japanese grid in 1996 and 97, respectively.

5. ISSUES INVOLVED IN NUCLEAR POWER EXPANSION

Rapid development of nuclear power in Asia raises a number of issues which are important to the power development of the countries concerned and which also have a global impact. These include; reactor safety, handling of radioactive waste including spent fuel, economic aspects, and nuclear non-proliferation. Naturally the shared interest over a wide range of issues makes world-wide co-operation indispensable.

5.1 Reactor Safety and Safety Culture

Safety issues are of universal importance. The future of nuclear power everywhere depends on the safe operation of all nuclear power plants, as evidenced, for example, by the impact of Chernobyl. Hans Blix, former director general of the IAEA, said once: "A nuclear accident anywhere is a nuclear accident everywhere". This suggests the possibilities of mutual co-operation on safety questions both within and outside the region. Several international institutions and programs are designed to promote nuclear safety.

Concern for safety pervades the entire nuclear power sector, in the design, manufacture and construction stages as well as operation. In design and engineering, this includes adopting sound basic designs, providing adequate margins and fail-safe arrangements and provision of proper measures for operation and maintenance. Standardization of reactor design also promotes safety and reliable operation. Advanced maintenance technologies reduce the risks of workers' exposure to radiation.

The countries of the region should continue to recognize the great importance of nuclear safety, strengthen the "safety culture", and adopt strict international standards of safety in plant performance including design, engineering and construction, operation and maintenance, and staff training at all levels. As safety is a universal issue, the Asian countries could benefit from regional exchange of information and data, and from adherence to many international programs and protocols governing reactor safety.

5.2 Economics of Nuclear Power

A basic question is whether nuclear power is cost-competitive with other forms of power generation. However, estimating comparative costs of nuclear and other base load power plants is a complex issue, and is subject to changes with time and also subject to considerable uncertainty. The cost of individual plants can vary widely according to location, design, siting requirements, and scale of plant. Many things can happen over the lifetime of a power plant that can impact the economics.

Operation and maintenance (O&M) costs in nuclear plants tend to rise as plants become older and regulatory requirements more strict. Some of these cost factors are also shared by fossil fuel plants, further complicating the comparisons. Fuel prices can change in unanticipated ways. Current low price of coal improves the competitiveness of coal-fired power generation. Lower price of uranium, which constitutes a very small part of nuclear power generation cost, does not have a big impact on total costs.

The sharp fall in the price of gas, its increased availability, and highly efficient technologies (the combined cycle gas turbine) have recently introduced a new competitive element into power generation choices.Advances in other non-nuclear technologies for power generation (renewable, advanced coal-burning technologies) can also have a major impact on the competitive economics of power generation.

Other changes with major implications for nuclear power in the past have been regulatory requirements involving expensive redesign and retrofitting, and licensing processes that have added years to construction and commissioning time. Such delays add nuclear power cost drastically as the capital cost of nuclear power plants represents an exceptionally high share of total costs. Moreover, the capital-intensive nuclear power plant is highly sensitive to the discount rates used to calculate their levelised cost.

Asian countries should work jointly with international organizations to develop acceptable models for determining the cost and economics of nuclear power, relative to their own circumstances. These models should provide consistent means of estimating the uncertainty range for each major cost component.

5.3 Management of Radioactive Waste and Spent Fuel

The safe storage of radioactive waste has proved to be one of the most intractable global problems of nuclear power generation, and one that could potentially inhibit further development of nuclear power in the future if not satisfactorily addressed. Discharge of spent fuel from world nuclear power plants was estimated at about 32,000 metric tons in 1997, and is expected to be reduced to about 10,000 tons per annum in the near future.

The current storage procedures for "once-through" fuel cycle concept are to temporarily store spent fuel assemblies at the reactor site in specially designed water-filled pools. Those pools were originally envisioned for short-term storage, but in the absence of reprocessing alternatives, and considering the difficulties in selecting permanent disposal sites, these are now providing longer-term storage.

As in other regions, waste management poses major problems in Asian countries. In Japan, each utility is responsible for storing the waste it produces, under government supervision. A permanent disposal facility for low-level waste is already in operation at Rokkasho-Mura. Longer-term solutions such as constructing a facility for vitrified waste or deep underground disposal are being investigated.

In Korea, waste management is the responsibility of the utility, KEPCO. Low-level waste is stored on site, as is spent fuel. At present, there are no tangible activities for storage of radwaste. High level waste, i.e., spent fuel, is stored on site, and no site has yet been selected for the deep underground disposal of high level waste. According to the mutual agreement, Taipower attempted to ship radwaste to North Korea, but this transaction has been hampered due to regional politico-diplomatic argument.

In India, each nuclear station stores its own waste. India follows a closed fuel cycle approach and has established fuel reprocessing plant and associated facilities for vitrifying and immobilizing high level waste. India operates an interim-storage facility with surveillance for high level waste, while ultimate deep underground disposal is being investigated. In Pakistan, intermediate- and high-level waste is currently stored at the associated nuclear facilities.

At present, none of the Asian countries is equipped with a permanent central facility for the disposal of high level waste. As many Asian governments are planning the expansion of nuclear power, the issue of nuclear waste management must receive continued priority attention. Given concerns in some countries, the possibility of regional storage facilities could be examined.

5.4 Nuclear Non-Proliferation

One of major issues in making use of civilian nuclear energy is the probability of its risk related to the diversion of nuclear materials and proliferation of nuclear weapons.

Asian nations committed to the use of nuclear power should take steps to the enhancement of regional co-operation in developing advanced reactors and in operating nuclear fuel cycle facilities; and all such endeavors should be implemented with crystal-clear transparency in sensitive nuclear materials management in strict compliance with the nuclear non-proliferation regime.

North Korea and its nuclear facilities present a special concern to the global village from the viewpoint of nuclear non-proliferation regime. North Korea became a party to the Treaty (NPT) in 1985 but delayed more than six years before agreeing, in April 1992, to permit IAEA inspection of its nuclear activities.During that interval, it is known to have produced a quantity of plutonium that may be sufficient for one-to-three nuclear weapons. As of March 1998, it has not satisfactorily accounted for this material and is not in compliance with the IAEA safeguards obligations under the Treaty because of its refusal to allow the IAEA to permit an IAEA "special inspection" of two nuclear waste sites believed to contain information regarding the production of plutonium in the past. Under an "Agreed Framework" signed with the United States in October 1994, North Korea has agreed to resolve these issues at a future date. In the meantime, it has accepted restrictions on its nuclear activities that go beyond its obligations under the NPT, including a freeze on the operation and construction of a number of sensitive facilities.

At about 40 km northwest of the Yonghyun nuclear complex, there has been the construction of a huge underground facility, and this suspicious facility is believed to be the site for North Korea's clandestine nuclear weapons production.

India and Pakistan have nuclear power plants and are seeking to expand their programs. But they are neither parties to the NPT nor have ratified the CTBT (Comprehensive Test Ban Treaty), and are refusing to accept the full-scope safeguards required by most supplier states. Under such circumstances, India detonated five nuclear devices at two days apart in May 11-13, 1998 at Rajastan desert where her first nuclear bomb test had taken place in 1974. Pakistan claimed to have detonated 5 bombs on 28th of May, two weeks after the India's tests. On May 30th, she conducted another test at the same site. The Indo-Pakistan's tests may become conducive to igniting chain reactions among the threshold countries that will see them as an open invitation to acquiring this dreadful technology. This cannot but be a worrisome concern to the global and Asian nuclear communities.

The Asian nuclear power industry and authorities should work with other nations to reaffirm their commitment to the non-proliferation regime. Areas for co-operation could include physical protection of nuclear materials, its control and accounting, regional co-operation in nuclear fuel cycle activities, and advanced reactor development. Asian governments should increase their efforts for transparency and undertake systematic confidence-building measures in order to contribute to minimizing proliferation concerns in and outside the region.

6. CONCLUSIONS AND RECOMMENDATIONS

The Asian region is likely to account for a large part of the total increase in nuclear capacity over the next few decades. These countries are motivated by the desire to expand the power for upgrading the standard of living and for expediting national modernization as well as enhancing energy security, conserving environment and benefiting from the technology spin-offs.

Although most of the emphasis in this report has been placed on the inter-co-operation within the region, there is also room for enhanced co-operation between the Asian countries and others outside the area. The benefit should be mutual and reciprocal. Some of the countries outside Asia may be able to pass on the benefits of their past experience to the Asian partners, but they will also be able to learn from the experience of rapid nuclear development in Asia, including evolutionary improvements to technology, at a time when there is virtually no new activity in their own countries.

In terms of reactor and fuel cycle technology development, the advanced western nuclear community resembles a fast-running rabbit, whereas the Asian partner is similar to a turtle. Now this Asian nuclear turtle is moving slowly but steadily towards the self-reliance of its technology and also towards the continued development of evolutionary water-cooled reactors and related fuel technology at a time when there is virtually no new nuclear projects in the western hemisphere and while the rabbit is taking a nap under a tree.

From the historical perspectives, it may be the Asian nuclear turtles who are assigned to keep this nuclear flame continuously burning and to fuel this nuclear candle for the future of mankind and for the world to be less pollutive.

Several decades ago, Rabindranath Tagore, the first Asian Nobel laureate from India, wrote a poem for Korea as follows ;

> In the golden age of Asia ,
> Korea was one of the lamp-bearers.
> And that lamp is waiting to be lighted once again
> For the illumination in the East.

I think I know what he meant by light and the lamp. This is nothing but light from the nuclear lamp. The light to be lit by the nuclear lamp which is another name of the evolutionary water-cooled reactor which has been developed by the Korean nuclear community under the direction of Korea Electric Power Corporation in the nomenclature of KNGR or Korea Next Generation Reactor. Thank you.

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KEY FACTORS IN THE DECISION-MAKING PROCESS

(Session II a)

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TECHNOLOGICAL READINESS OF EVOLUTIONARY WATER COOLED REACTORS

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Abstract

Nuclear energy has evolved to a mature industry that supplies over 16% of the world's electricity, and it represents an important option for meeting the global energy demands of the coming century in an environmentally acceptable manner. New, evolutionary water cooled reactor designs that build on successful performance of predecessors have been developed; these designs have generally been guided by wishes to reduce cost, to improve availability and reliability, and to meet increasingly stringent safety objectives. These three aspects are important factors in what has been called technological readiness for an expanded deployment of nuclear power; a major increase in utilization of nuclear power will only occur if it is economically competitive, and meets safety expectations. To this end, the industry will also have to maintain or improve the public perception of nuclear power as a benign, economical and reliable energy source.

1. INTRODUCTION

During the last 50 years, nuclear energy has evolved from the research and development environment to an industry that supplies over 16% of the world's electricity. There is now an opportunity for nuclear energy to significantly contribute to the global energy demands of the coming century in an environmentally acceptable manner. At the end of 1997, according to data reported in the Power Reactor Information System, PRIS, of the IAEA, there were 437 nuclear power plants in operation and 36 under construction. Over eight thousand five hundred reactor-years of operating experience had been accumulated. Of the operating plants, 346 were LWRs totalling 306 GWe and 30 were HWRs totalling 16.4 GWe. The considerable experience and lessons learned from these plants are being incorporated into new water cooled reactor designs.

In the early years of nuclear energy utilization, various design concepts were developed and implemented. Subsequently, a pattern of evolutionary improvements led to the successful designs of today. Now, most of the efforts in water cooled reactor development are on evolutionary designs. This arises from conservatism in licensing new developments; from the conservatism of utilities which recognise that financial risks can be better controlled if new developments build on proven technologies; and from the reluctance of governments to contribute to the financing of new prototypes.

Evolutionary developments have generally progressed to reduce costs, to meet increasing safety demands and to increase reliability and availability. The nuclear industry is now faced with increased competition from natural gas, the trend to deregulation of electricity markets, difficulties in financing new projects, and increasingly stringent safety objectives. This paper addresses the technology readiness of evolutionary developments in water-cooled reactors for meeting these challenges.

2. EVOLUTIONARY DESIGNS

Advanced nuclear power plant designs or concepts can be divided into two categories: evolutionary designs and innovative designs that require substantial development efforts. A natural dividing line between these two categories arises from the necessity of having to build and operate a prototype or demonstration plant to bring a concept with much innovation to commercial maturity, since building and operating such a plant represents a significant step increase, with respect to both cost and

Advanced design

Different types of new nuclear plants are being developed today that are generally called advanced reactors. In general, an advanced plant design is a design of current interest for which improvement over its predecessors and/or existing designs is expected. Advanced designs consist of evolutionary designs and designs requiring substantial development efforts. The latter can range from moderate modifications of existing designs to entirely new design concepts. They differ from evolutionary designs in that a prototype or a demonstration plant is required, or that insufficient work has been done to establish whether such a plant is required.

Evolutionary design

An evolutionary design is an advanced design that achieves improvements over existing designs through small to moderate modifications, with a strong emphasis on maintaining proven design features to minimise technological risks. The development of an evolutionary design requires at most engineering and confirmatory testing.

Innovative design

An innovative design is an advanced design which incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice. Substantial R&D, feasibility tests, and a prototype or demonstration plant are probably required.

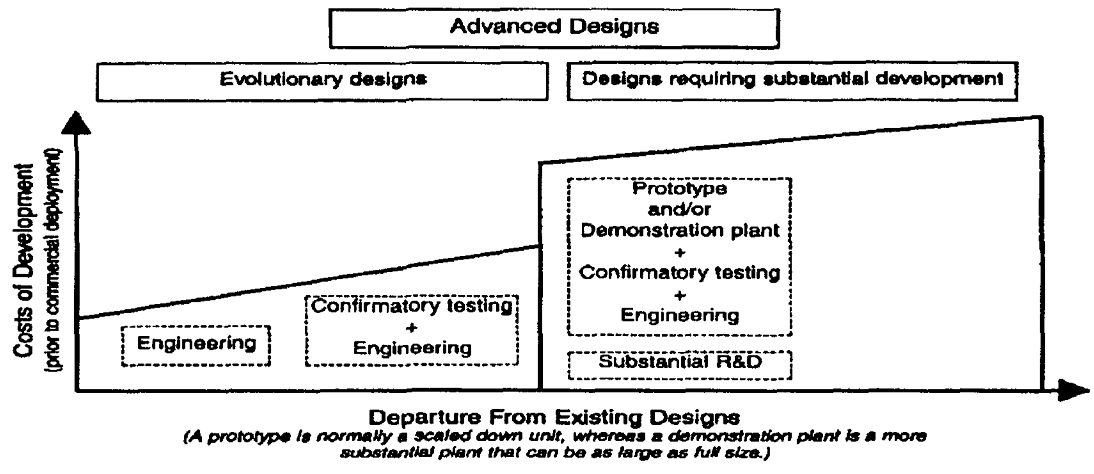

Fig. 1. Efforts and development costs for advanced designs versus departure from existing designs (Terms are excerpted from Ref. 1).

time, in the resources needed for the development. Designs in both categories need engineering, and may also need research and development (R&D) and confirmatory testing prior to finalizing the design of either the first plant of a given line in the evolutionary category, or of the prototype and/or demonstration plant for the second category. The amount of such R&D and confirmatory testing depends on the degree of both the innovation to be introduced and the related work already done, or the experience that can be built upon. This is particularly true for designs in the second category where it is entirely possible that all a concept needs is a demonstration plant, if development and confirmatory testing is essentially completed. At the other extreme, R&D, feasibility tests, confirmatory testing, and a prototype and/or demonstration plant are needed in addition to engineering. Different tasks have to be accomplished and their corresponding costs in qualitative terms are a function of the degree of departure from existing designs. In particular, a step increase in cost arises from the need to build a reactor as part of the development programme (see Figure 1).

3. TECHNOLOGY READINESS

Technology readiness relates to many of the factors that influence the success of new designs. Currently-operating plants must continue to operate safely and reliably, and new designs must be able to economically meet increasingly stringent safety objectives. Economic competitiveness is a key to the introduction of any new product. Technologies must be available for reducing the investment cost, fuel cycle costs and operating costs. Readiness is not only based on what has been designed, fabricated and tested, but on the continuing presence and vigour of the institutional framework and technological base, including R&D and design organizations, manufacturing and construction resources, strong yet balanced and responsive regulatory organizations, and the infrastructure that trains and maintains the necessary human resources. During this Symposium, these factors will be addressed in detail, and the design organizations will describe their most significant technological developments, the status of their validation, how these developments meet regulatory and utility requirements, and projected costs. This Symposium is a major activity within the IAEA's nuclear power programme which promotes information exchange and co-operative research in reactor development, and provides a source of balanced, objective information for all Member States on the current status and recent advances in reactor design and technology [2, 3, 4, 5, 6, 7, 8].

4. DEVELOPMENT OF NEW DESIGNS

The technology of evolutionary water cooled reactors is building on the growing experience base and results of R&D programmes. World-wide, considerable efforts are being made to develop advanced nuclear power plants to meet the future demand for energy. Various organizations are involved, including governments, industries, utilities, universities, national laboratories, and research institutes. Expenditures for development of new designs, technology improvements, and the related research for all major reactor types combined is estimated to exceed US $ 2 billion per year.

For water cooled reactors, utility requirements documents have been formulated to guide the design and development activities by incorporating experience from current plants with the aim of reducing costs and licensing uncertainties by establishing a technical foundation for the advanced designs. Large evolutionary designs are being developed with power outputs up to the 1500 MWe range which incorporate mainly proven, active engineered systems to accomplish safety functions, and mid-size evolutionary designs are being developed which place more emphasis on utilization of passive safety systems. The experience base on which these developments are building is large: over 6100 reactor years for LWRs and over 600 reactor years for HWRs. Common goals for these new designs are high availability, user-friendly features, competitive economics and compliance with internationally recognized safety targets.

5. MEETING INCREASINGLY STRINGENT SAFETY OBJECTIVES

As the number of nuclear plants world-wide increases, safety targets are becoming more stringent. Operational safety records are good and steadily improving as shown by the low number of either unanticipated trips or spurious actuation of engineered safety systems. Thus, the basic safety level is considered acceptable, and the focus of the increasingly stringent safety objectives for new designs is on very low probability accident scenarios involving severe core damage. INSAG-3 [9] notes that "The target for existing nuclear power plants is a likelihood of occurrence of severe core damage that is below about 10^{-4} events per plant operating year. Implementation of all safety principles at future plants should lead to the achievement of an improved goal of not more than about 10^{-5} such events per plant operating year. Severe accident management and mitigation measures should reduce by a factor of at least ten the probability of large off-site releases requiring short term off-site response." The more stringent safety target for future plants was confirmed by INSAG-5 [10] which notes that [evolutionary] light and heavy water nuclear plants should meet the long term target of a level of safety ten times higher than that of existing plants. INSAG-10 [11] notes that prevention of accidents remains the highest priority among the safety provisions for future plants and that probabilities for severe core damage below 10^{-5} per plant year ought to be achievable. However, values that are much smaller than this would, it is generally assumed, be difficult to validate by methods and with operating experience currently available. Improved mitigation is therefore an essential complementary means to ensure public safety.

Evolutionary water-cooled nuclear power plants incorporate various technical features to meet the safety targets. In many cases, these features have been tested to demonstrate technological readiness. Examples of such features are:

- increased margins and grace periods, (e.g., larger water inventories, large pressurizer, large steam generators, lower power densities, negative reactivity coefficients) to limit system challenges;
- redundant and diverse safety systems to perform simplified tasks, improved physical separation between systems, and utilization of components of proven high reliability;
- reliable depressurization systems to preclude high pressure core melt sequences;
- passive cooling and condensing systems;
- provision for corium confinement and cooling;
- containments large enough to withstand the pressure and temperature from design basis accidents without fast acting pressure reduction systems, sometimes surrounded by a second containment which provides protection against external missiles and allows for detection and filtration of activity leaking from the first containment;
- systems to control hydrogen concentrations during accidents.

Importantly, design measures both for increased prevention as well as for accident mitigation tend to increase capital cost, although preventive measures may provide higher plant availability and therefore have a positive cost component. The added costs for measures only aimed at mitigating accidents must be overcome by other savings.

6. TECHNOLOGY FOR ENHANCED COMPETITIVENESS

Despite the prevailing low fossil-fuel prices, the generating cost of nuclear electricity continues to be competitive with fossil fuel for base-load electricity generation in many countries. Although the large capital investment required for nuclear power plants is a disadvantage, especially in developing countries, the nuclear fuel cycle cost is relatively low. Moreover, the prices of fossil fuels are likely to increase over the long term because the resource is limited and also if pressures are applied - by political or financial instruments, to discourage use; and there is still scope in the nuclear industry for standardisation, modular construction, shorter construction periods, higher burnup and simplification, resulting in better performance and lower generation cost.

In the next few years, however, nuclear utilities will experience an operating environment in which nuclear power plants will face increased competition, in a deregulated energy market, with other suppliers of electricity. Data on operating costs will be analysed to determine whether the continued operation of nuclear power plants provides power to consumers at the least cost. This competitive environment has significant implications for plant operations, including efficient use of all resources; more effective management of plant activities, such as outages and maintenance; and sharing of resources, facilities and services among utilities.

Achieving high plant availability and reliability are essential factors for achieving good economics, and they require attention to both the nuclear island and the balance of plant. Nuclear power plants world-wide are showing a steady increase in the average energy availability factor, which has increased from approximately 70 percent in 1989 to 77.4 percent[1] in 1997, with some utilities achieving significantly higher values. This is being achieved through integrated programmes including personnel training, quality assurance, improved maintenance planning, as well as technological advances in plant

[1] Based on IAEA Power Reactor Information System (PRIS) data. In PRIS, the energy availability factor is defined as 100 [1-EL/E_m] with E_m being the net electrical energy which would have been produced at maximum capacity under continuous operation during the reference period, and EL is the electrical energy which could have been produced during the reference period by the unavailable capacity. (The numbers reported here are for plants with capacity greater than 100 MW(e) and with more than one year of commercial operation).

components and systems, and in inspection and maintenance techniques. International co-operation is playing a key role in this success. The various programmes of the World Association of Nuclear Operators (WANO) to exchange information and encourage communication of experience, and the activities of the IAEA including projects in nuclear power plant performance assessment and feedback, effective quality management, and information exchange meetings on technology advances, are important examples of international co-operation to improve the performance of current plants.

Improved performance at current plants is supported by better application of existing technologies, for example, for processing information regarding the condition of components, and performing surveillance and diagnostics. Activities are being implemented at current plants to analyse information from operation of components and systems to understand the causes of unavailability, and to improve work processes during maintenance. New technologies are also being developed with the aim of improving performance of current plants. Examples include development of high burnup fuel which supports longer cycle length, computer-aided systems to provide early indication of sensor or component degradation, and simpler systems for control of hydrogen during accident conditions (systems that require considerably less testing and maintenance and thereby reduce outage duration).

Significant improvements are also being achieved in primary system components, which will contribute to high availability. As an example, the dominant cause of damage to PWR steam generator tubing has been due to corrosion of the tubing and support structures. Large efforts in several countries have been and are being carried out to control and improve the service environment to extend the service life of steam generator tubes. New materials (e.g. Inconel 690) have been shown to have superior corrosion resistance compared to Inconel 600 and are now used for new and replacement PWR steam generators.

Enhanced utilization of forgings in the pressure retaining parts of the primary system, to reduce the number and length of welds, is another example of efforts that aim at facilitating and improving plant operation. Weld length reductions reduce ISI (In-Service Inspection) requirements and work to be done in high-radiation areas, leading to less occupational radiation exposure.

User requirements documents for future plants specify plant availability factors of 87% and above. For new plants, the basis for achieving high performance is being laid down during the design phase. For example, design for short outages, design for on-line maintenance, and design for increased margins along with an overall goal of simplicity should contribute to the improved level of availability requested by user requirements documents. Also, advances such as better man-machine interface using computers and improved information displays, greater plant standardization, improved maintainability, and better operator qualification and simulator training, which have been applied at current plants will contribute to high performance of future plants.

Improved nuclear plant economics may also be gained by widened design margins, provided that the associated cost increase can be outweighed by a gain in operational availability. Substantial design margins provide benefit by:

- providing capability to accommodate disturbances and transients without causing challenges to the plant safety, and initiation of engineered safety systems;
- providing margin to enhance system and component reliability, and to minimise the potential of exceeding specified limits which would require de-rating or shutdown;
- providing additional assurance that the longer plant life requirement of 60 years can be met.

A significant thermal margin for the core design serves to ensure for the utility that the plant will be capable of being operated at 100% power when started up, in spite of unforeseen material and/or design problems for the fuel; in addition, such margins yield specific benefits with respect to enhanced transient performance and increased operational flexibility. In the longer term, the design margin also provides an option that may become attractive for the utility once experience has been gained with the new plant: uprating of the plant power level at a marginal cost that will be small compared with adding other new capacity.

7. CONCLUSION

In summary, evolutionary designs are being developed with the objectives of reduced costs, and higher availability while meeting increasingly stringent safety targets. The designs incorporate evolutionary improvements, and features which are well supported by operating experience and/or research, development and confirmatory testing. These new designs are ready to ensure that nuclear energy can continue to play an important and increasing role in global energy supply.

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ECONOMIC COMPETITIVENESS REQUIREMENTS FOR EVOLUTIONARY WATER COOLED REACTORS

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Abstract

This paper analyses the necessary economic conditions for evolutionary water cooled reactors to be competitive. Utilising recent national cost data for fossil-fired base load plants expected to be commissioned by 2005-2010, target costs for nuclear power plants are discussed. Factors that could contribute to the achievement of those targets by evolutionary water cooled reactors are addressed. The feed-back from experience acquired in implementing nuclear programmes is illustrated by some examples from France and the Republic of Korea. The paper discusses the impacts on nuclear power competitiveness of globalisation and deregulation of the electricity market and privatisation of the electricity sector. In addition, issues related to external cost internalisation are considered.

1. INTRODUCTION

Economic competitiveness is a cornerstone for the successful deployment of any electricity generation source and technology. Although decisions on technologies and energy mixes for electricity generation have to take into account a variety of non-economic issues, including social, environmental and health impacts, decisions taken by utilities are based primarily on the costs of generating electricity from alternative energy sources and technologies available on the market. Therefore, designers and manufacturers of evolutionary water cooled reactors must produce plants whose costs are competitive with other options.

While assessing the competitiveness of alternative sources, the evolution of the policy-making framework for the electricity sector should be taken into account. This evolution creates new challenges and opportunities for different generation technologies, including nuclear power. Deregulation of the electricity market and privatisation of the sector are changing the criteria upon which assessments of competitiveness are based. Private investors will tend to prefer low capital intensive technologies that offer a rapid return on investments. Market deregulation poses challenges for capital intensive technologies, such as nuclear power, because the resulting open competition for supplying electricity will introduce a higher uncertainty on the level of sales by each producer. In order to reduce financial risks, producers will tend to seek more flexible generation strategies that are based upon small size power plants with relatively low investment costs and short pay-back times. Nuclear power will be challenged to retain its competitive position in such a market, owing to the fact that it is a relatively complex technology that requires sophisticated industrial and R&D infrastructures which might be difficult for the private sector to support. On the other hand, the reduction of barriers to bulk electricity exchange via extended networks offers new market opportunities for large units that have stable long term generation costs, such as nuclear power plants.

The increasing awareness of environmental issues and the recognition of broad macroeconomic and social effects arising from technology choices are leading to new approaches and additional criteria in the comparative assessment of different generation options. Cost comparisons of generation technologies can be taken beyond the traditional approach of calculating the direct economic costs to the utility by internalising other costs to society (externalities) insofar as feasible. Internalising externalities might enhance the competitiveness of nuclear power versus coal and gas-fired power plants. Owing to the early recognition of the need to adequately protect the public and environment from ionising radiation, the classic levelised cost assessment already takes into account most of the elements related to health and environmental impacts of nuclear power generation, from mining through electricity generation to decommissioning of the facilities, waste management and disposal. Also, the costs related to the application of safety standards and regulations are embedded in the investment, operation and maintenance costs of nuclear power plants. On the other hand, the liabilities arising from fossil fuel electricity generation (for example, the potential costs of greenhouse gas emissions) are not taken fully into account at present, and their inclusion would increase the costs of fossil fuel based generation relative to nuclear.

This paper addresses the cost economics necessary for evolutionary water cooled reactors to be competitive. Target costs that would allow evolutionary water cooled reactors to compete favourably with alternatives are identified. Factors affecting nuclear power costs are analysed and lessons learnt from past experience are illustrated by examples of cost reduction achieved through the successful implementation of nuclear power programmes in France and the Republic of Korea. The challenges and opportunities resulting from the new economic landscape, and the ways in which they might affect the competitiveness of evolutionary water cooled reactors are discussed. Also, external costs of nuclear power and alternative generation sources are highlighted.

2. TARGET COSTS FOR EVOLUTIONARY WATER COOLED REACTORS

The evolutionary water cooled reactors that are currently being developed will be commercially available for commissioning by 2005-2010. They will compete essentially with state-of-the-art coal-fired and gas-fired power plants. Renewable sources, besides hydropower, generally are not considered for base load generation. For example, in the last OECD study on projected costs of generating base-load electricity [1], participating countries provided cost estimates mainly for coal-fired, gas-fired and nuclear plants.

The data on projected generation costs provided by countries participating in the OECD study show that those costs vary from country to country for similar technologies owing to their specific economic context. However, the levelised lifetime generation costs obtained in the study, using a common framework and generic assumptions, provide a consistent basis for assessing future generation costs at the conceptual level. Cost estimates were provided by participating countries expressed in their national currencies of 1 July 1996; for consistency sake, cost estimate results were converted to dollars of the United States as of 1 July 1996, using the official exchange rate applicable at that date.

On average, projected generation costs for coal-fired power plants are around 41 mill/kWh [1] at 5 per cent per annum real discount rate and around 51 mill/kWh at 10 per cent per annum discount rate. Those costs are based upon coal prices ranging from 1 $/GJ to 2.8 $/GJ in 2005 - year of commissioning of the plant - and increasing at an average escalation rate of 0.3 per cent per annum. For gas-fired power plants, the average projected generating costs are 40 mill/kWh and 43 mill/kWh at 5 per cent and 10 per cent discount rate, respectively. The gas prices assumed vary between 1.6 $/GJ and 5.4 $/GJ in 2005 with a 0.8 per cent per annum average escalation rate. In the same study, the average generation costs for nuclear power plants are 32 mill/kWh and 49 mill/kWh, respectively at 5 per cent and 10 per cent discount rate.

Table I. Projected levelised generation costs (1996USmill/kWh)

	At 5 per cent discount rate			At 10 per cent discount rate		
Country	Coal	Gas	Nuclear	Coal	Gas	Nuclear
Canada	29.2	30.0	24.7	37.0	33.0	39.6
Finland	31.8	35.9	37.3	39.1	41.1	55.9
France	46.4	47.4	32.2	59.5	53.3	49.2
Korea	34.4	42.5	30.7	45.0	47.0	48.3
Spain	42.2	47.9	41.0	54.7	54.4	63.8
Turkey	39.8	30.7	32.8	48.7	33.9	51.8
United States	25.0	23.3	33.3	34.7	23.6	46.2
Brazil	35.4	28.5	33.1	43.2	32.7	46.7
China	31.8	n.a.	25.4	40.0	n.a.	39.0
India	33.0	n.a.	32.8	40.2	n.a.	51.0
Russia	46.3	35.4	26.9	55.3	39.0	46.5

Table I summarises the results obtained in the OECD study for the reference cases, i.e., 40 year lifetime and 75 per cent load factor, in the eleven countries that provided cost data for nuclear power and at least one alternative. All those countries, except China and India which did not provide data for gas-fired plants, reported cost estimates for coal-fired, gas-fired and nuclear power plants. In cases of multiple plant submissions in the study, only the cheapest power plant for each technology/fuel is shown.

At 5 per cent discount rate, levelised generation costs for coal-fired power plants range from 25 to 46 mill/kWh, the average value being around 36 USmill/kWh; for gas-fired power plants, the range is 23 to 48 USmill/kWh, the average value being around 36 USmill/kWh; and for nuclear power plants, the range is 25 to 41 USmill/kWh and the average value is around 32 USmill/kWh. At 10 per cent discount rate, levelised cost ranges are 35 to 60 USmill/kWh, 24 to 54 USmill/kWh and 39 to 64 USmill/kWh for coal-fired, gas-fired and nuclear power plants, respectively, with the average values around 45, 40, and 49 USmill/kWh, respectively.

In the eleven countries where coal and nuclear options are considered, the ratios between projected costs of nuclear and coal generated electricity range from 0.58 to 1.33 at 5 per cent discount rate and from 0.83 to 1.43 at 10 per cent discount rate. In the ten countries where gas and nuclear options are considered, the ratios between projected costs of nuclear and gas generated electricity range from 0.68 to 1.43 at 5 per cent discount rate and from 0.92 to 1.96 at 10 per cent discount rate. In the same countries, the ratios between projected costs of coal and gas generated electricity range from 0.76 and 1.24 at 5 per cent discount rate and from 0.68 and 1.04 at 10 per cent discount rate.

As indicated above, the ranges of generation costs for each technology/energy source are quite broad, underscoring the observation that competitiveness should be assessed on a case by case basis at the country and utility level, based upon specific technical and economic conditions applicable in each case. Nevertheless, the average generation costs given above provide an indication of target costs in order for evolutionary water cooled reactors to be competitive.

Owing to uncertainties on projected cost elements and to the conceptual level of detail inherent within international studies based upon generic assumptions, small differences in generation costs are not significant. Differences higher than 10 per cent, however, may be considered indicative of the relative competitiveness of alternative options in each country. Within the eleven countries that provided data for nuclear power and at least one other option, at 5 per cent discount rate, nuclear is the cheapest by a margin of at least 10 per cent in five countries, coal is the cheapest by a margin of at least 10 per cent in one country and gas is the cheapest by a margin of at least 10 per cent in one country. At 10 per cent discount rate, nuclear is the cheapest option by a margin of at least 10 per cent in no country, coal is cheapest option by a margin of at least 10 per cent in one country and gas is the cheapest option by a margin of at least 10 per cent in five countries. As would be expected, as the discount rate increases, technologies with lower capital intensity fare better.

3. KEY FACTORS FOR ENHANCING EVOLUTIONARY WATER COOLED REACTOR COMPETITIVENESS

The historical cost experience of nuclear power has been quite varied. In some countries, nuclear power has become the primary source of economic baseload generation. In other countries, particularly in the United States, the cost results of nuclear power have been inconsistent, with some facilities producing low cost electricity and other facilities closed before the end of their design life due to economic non-competitiveness. The causes of this inconsistency are beyond the scope of this paper, but the lessons learned from these events are very relevant to the future of nuclear power.

A number of studies have been undertaken to "learn from the past" and improve the economics of nuclear power. In 1990, the NEA investigated means to reduce capital costs of nuclear power used in different Member countries [2]. In the United States, the Electric Power Research Institute (EPRI), in co-operation with the utility industry and the U.S. Department of Energy, has developed a set of utility design requirements for next generation nuclear plants [3]. The EPRI Utility Requirements Document (URD) consists of a comprehensive set of design requirements for future LWRs. As a part of the URD, several key elements that are necessary to achieve deployment success have been defined. Many of these factors bear heavily on the potential economic competitiveness of new plants and are presented below.

Simplification – Nuclear power is, inherently, a relatively complex undertaking. Nevertheless, plant designs can seek to minimise the number of systems, valves, pumps, etc., consistent with essential functional requirements. A particularly important aspect of simplification is with respect to plant operations. Improved man-machine interfaces, simplified protective logic and actuation systems, and system designs which minimise operator demands will lead to higher plant availability and less accident risk.

Regulatory Stability – The requirements necessary to obtain regulatory approval must be clear and stable if costly redesign and plant modifications are to be avoided. In the United States, the Nuclear Regulatory Commission has developed rules to further define and improve the licensing process through a "one-step licensing process". Early dialogue between plant designers and regulatory bodies can lead to improved understanding of the regulatory requirements and how the plant design can meet them.

Standardisation – A key mechanism to achieve low costs is designing and constructing a standardised plant. A standardised approach creates efficiencies in engineering, construction, and schedule. With a standardised plant, the bulk of design and engineering activities can be performed once and the costs amortised over many units, licensing time and costs will be reduced, and construction techniques will become more refined, reducing both cost and schedule. Operation and maintenance costs will be reduced through reduced operator error and improved maintenance efficiencies.

Improved Constructibility – By incorporating the items identified above, the construction duration will be kept to a minimum, which will greatly influence the plant capital cost. In addition to the above items, improvements in construction will also be made when there is a large fraction of the design complete before construction begins. As an example, the EPRI URD requires that ninety percent of the design be complete (i.e., 90 percent of design drawings must be 100 percent complete) before plant construction starts. Another aspect of improved constructibility relates to modularisation, whereby segments of a plant can be fabricated and assembled in a factory environment and shipped to the site for placement and interconnection. The factory setting provides a better work environment as activities tend to have greater quality control with the potential for greater automation and higher productivity. Modular construction also reduces site congestion and improves construction schedule as module production can take place away from and in parallel with site specific activities.

An additional factor that can influence economic competitiveness is the use of multiple unit sites. In addition to the sharing of the site land cost, site licensing costs can also be shared among multiple units. During the construction phase, considerable efficiencies and associated savings can be gained from phased construction and rolling the various craft teams from one unit to the next. Also, by construction repetition, there is craft labour learning that reduces the time to perform a given task and correspondingly reduces both construction labour cost and schedule. Finally, common facilities such as administration and maintenance buildings, warehouses, roads, and guard stations can be shared by multiple units at a common site.

4. FEED-BACK FROM EXPERIENCE

As over half of the total generation cost of a nuclear plant is related to capital cost, reducing the plant capital cost is a prerequisite for enhancing the competitiveness of nuclear power. As mentioned in the previous section, a study carried out by the NEA in 1990 **[Error! Bookmark not defined.]** analysed means to reduce the capital cost of nuclear stations, identifying as the most significant: plant size, multiple unit sites, design improvement, standardisation, modularisation and performance improvement. A second study, to be completed by the end of 1998, is revisiting the issue drawing from experience acquired in Member countries.

The French experience is of interest in this connection since its large nuclear power programme, based upon standardised units and large series orders, led to competitive nuclear generation costs as compared with fossil fuels. The impacts of unit size and number of units constructed on the same site, according to French data, are illustrated in Table II.

Also, in the French case, the effect of series order is estimated to have been significant. The "first-of-a-kind" (FOAK) initial cost may be between 15 per cent and 55 per cent higher than the cost of a series unit depending on the differences between a new design and previous reactors. When a series of reactors is ordered, additional cost reductions resulting from productivity effects are possible from the third unit on. With a 2 per cent productivity gain for each new unit after the second one, the capital cost of the eighth unit in the series is 10 per cent lower than the capital cost of the first unit.

The Korean nuclear power programme is characterised by standardisation and technology self-reliance. Since 1987, Korea has developed the Korea Standard Nuclear Power Plant (KSNP), a 1000 MWe PWR. Today, one KSNP unit is in operation and five more units are under construction. The capital costs of subsequent KSNPs, based on contracts, are illustrated in Table III.

The most noticeable cost reductions were achieved in Nuclear Steam Supply System (NSSS) equipment, turbine plant equipment, and design and engineering. The cost reduction in NSSS and turbine equipment results from enhanced technology self-reliance and increased productivity in manufacturing through construction repetition and design evolution. The largest beneficiary of

Table II. Overnight costs of nuclear power plants, normalised to 1.0 for 1 x 1000 MWe unit

1 x 300*	2 x 300*	1 x 650*	2 x 650	**1 x 1 000***	2 x 1 000	1 x 1 350	2 x 1 350*
1.82	1.44	1.22	1.0	**1.0**	0.84	0.87	0.75

* *Reactor size in MWe*

Table III. Capital costs of subsequent KSNPs, normalised to 1.0 for 1st & 2nd units

	1st & 2nd units	3rd & 4th units	5th & 6th units
Direct cost	1.0	0.9	0.9
Indirect cost	1.0	0.9	0.73
Contingency	1.0	0.9	0.85
Total capital cost	1.0	0.9	0.85

standardisation could be design and engineering costs. Another factor that influenced the cost reduction significantly is phased construction of continuing projects.

In addition to KSNP, Korea has started a programme for the development of the Korea Next Generation Reactor (KNGR), a 1 300 MWe PWR. The key objective of the KNGR development programme is to enhance safety and economics. The expected cost reduction as compared with existing KSNP and influencing factors were estimated in an economic viability study on KNGR, the results of which are shown in Table IV.

5. ADAPTATION TO THE NEW ECONOMIC LANDSCAPE

Deregulation of the electricity market and privatisation of the sector are changing the criteria upon which assessments of competitiveness are based. Private investors will tend to prefer low capital intensive technologies that offer a rapid return on investments. Market deregulation poses challenges for capital intensive technologies, such as nuclear power, because the resulting open competition for supplying electricity will introduce a higher uncertainty on the level of sales by each producer.

To reduce financial risks, producers will tend to seek more flexible generation strategies that are based upon small size power plants with relatively low investment costs and short pay-back times. In order to retain a competitive position in such a market, designers of nuclear power plants should aim towards streamlined concepts, requiring less sophisticated industrial and R&D infrastructures, including consideration of smaller size modular units more adapted to uncertainties on future demand.

Competitive markets may raise the discount rate used by electric utilities because commercial risks for utilities will increase. Nuclear power plants that have high up-front capital costs would, thereby, have a handicap as compared with low capital intensive gas-fired units.

However, the reduction of barriers to bulk electricity exchange via extended grid networks offers new market opportunities for large units that have stable long term generation costs, such as evolutionary light water reactors.

Table IV. Expected capital cost reduction compared with KSNP and influencing factors

Influencing factor	Expected cost reduction
Standardised design	4.9%
Simplified design	> 4%
Capacity upgrade	8%
Reduced construction period	4%
Total capital cost reduction	> 16.9%

6. EXTERNAL COSTS

Beyond, direct levelised generation costs, as described above, there are external costs or benefits related to the production of electricity that are not directly borne by producers or consumers. Costs and benefits to society that generally are not incorporated in direct costs of electricity generation include: macro-economic impacts such as job creation, price stability and balance of payments; strategic factors such as security and diversity of supply; resource management and sustainability; and health and environmental impacts of residual emissions.

Ensuring diversity of supply and preserving energy security are undoubtedly of concern to policy makers. However, empirical and analytical studies aiming at assessing the value of energy diversity and security differ enormously in their results and conclusions that depend largely on country specific contexts. It is clear that nuclear power introduces diversity and reinforces security of energy supply, but the extent to which this might be reflected in generation costs and prices is questionable.

Environmental impacts are potentially the most significant external cost of electricity generation. The increasing awareness of global impacts on the environment and the broad acceptance of the concept of sustainability are leading analysts and decision makers to aim towards incorporating these parameters either explicitly or implicitly in the comparative assessment process.

Since nuclear power plants of the current generation already have very low external costs related to both normal operation and off-normal situations, greater internalising of externalities might enhance the competitiveness of nuclear power relative to coal- and gas-fired power plants.

A study carried out under the auspices of the European Commission concluded that external costs for fuel cycles are essentially country/technology specific and that large uncertainties on physical damages and their monetary values are large enough to make comparisons between alternative generation options very sensitive to local conditions and assumptions made in estimating those costs. However, the external cost ranges derived from the study (cf. Table V) indicate clearly that nuclear power externalities are lower than those of coal or gas, primarily due to the greater environmental emissions of fossil-fired plants.

The assessment of the external costs related to the operation of a nuclear reactor is based on monetary valuation of the associated health impacts, for both public and workers. Concerning the public, these impacts are associated with the radioactive releases from the nuclear power plant into the environment. Occupational health impacts include potential health effects of ionising radiation exposure as well as occupational accidents.

For a current 1 300 MWe French reactor, the cost associated with the health impacts of the electricity generation routine operation varies between 0.017 and 0.04 mill/kWh for a 3 per cent discount rate, depending on the site, with an average of 0.022 mill/kWh (without discounting, this cost reaches 0.57 mill/kWh, owing to long term impacts) [4]. This cost can be compared to the monetary

Table V. External cost estimates for electricity generation technologies*

Source	Lower estimate	Higher estimate
Coal	7	60
Gas	3	14
Nuclear	1	3
Biomass	1	12

* *Normalised to 1 for nuclear power lower estimate. [Source:* [1].*]*

value of 0.026 mill/kWh reported for a 900 MWe French PWR, which represents an older generation of reactors. For normal operations, it is important to note that, despite the differences existing in the characteristics of the sites and the releases, the differences between the two kinds of PWR are essentially insignificant. Moreover, if the construction and the decommissioning impacts are taken into account, the average value becomes 0.08 mill/kWh for both PWR types.

7. CONCLUDING REMARKS

Nuclear power is a capital intensive technology. As such, owner risks are greater for this technology in a changing or unstable business environment than for a technology with lower capital cost. With the global utility industry undergoing deregulation and privatisation, the market is not the most attractive for highly capital intensive technologies. In a free market, relative cost will have a great influence in the selection process. Evolutionary water cooled reactors must, therefore, project a very competitive total cost in order to be successfully deployed. In terms of broad application, this may be difficult to accomplish globally as fossil-fuel prices are currently low and are projected to rise only modestly. However, there are, and will continue to be, markets where indigenous fossil fuel supplies are scarce, and in these areas, nuclear power may have the economic advantage.

Nuclear power plant costs can be minimised if certain conditions exist at the outset. These conditions include design simplification and standardisation, clear and stable regulatory requirements, a high fraction of design completion before construction, use of multiple unit sites with phased construction, and use of modular construction.

A factor that may influence the competitiveness of nuclear power in the future is externality costs. National policy issues of energy security and diversity of supply may modify the selection process from one of pure relative internal cost. In addition, the global environmental impacts of various power generation technologies are not completely internalised at present. Should this occur, nuclear power will likely have an improved economic ranking worldwide. It is not clear, however, when and if such recognition will take place. Therefore, evolutionary water cooled reactors must be competitive at this time solely on the basis of their direct, internalised costs. This paper has identified some of the factors necessary for this to occur.

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THE CHALLENGE OF FINANCING NUCLEAR POWER PLANTS

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Abstract

To date, more then 500 nuclear power reactors have been successfully financed and built. Experience in recent nuclear projects confirms that nuclear power will not cease to be a viable option due to a worldwide financing constraint. For financing nuclear plants there are special considerations : large investment; long lead and construction times; complex technology; regulatory risk and political risk. The principal preconditions to financing are a national policy supporting nuclear power; creditworthiness; economic competitiveness; project feasibility; assurance of adequate revenues; acceptability of risks; and no open-ended liabilities. Generally, nuclear power plants are financed conventionally through multi-sources, where a package covers the entire cost . The first source, the investor/owner/operator responsible for building and operating the plant, should cover a sizable portion of the overall investment. In addition, bond issues, domestic bank credits etc. and, in case of State-owned or controlled enterprises, donations and credits from public entities or the governmental budget, should complete the financing . A financially sound utility should be able to meet this challenge. For importing technology, bids are invited. Export credits should form the basis of foreign financing, because these have favorable terms and conditions. Suppliers from several countries may join in a consortium subdividing the scope of supply and involve several Export Credit Agencies (ECAs). There are also innovative financing approaches that could be applied to nuclear projects. Evolutionary Reactors with smaller overall investment, shorter construction times, reliance on proven technology, together with predictable regulatory regimes and reliable long-term national policies favorable to nuclear power, should make it easier to meet the future challenges of financing.

1. THE FINANCING ISSUE

All industrial projects requiring investment capital must be financed and Nuclear Power Plants are no exception. Financing is an issue that grows in importance in proportion to the capital required and the risks involved. Solving the financing issue is a challenge to the investor; if adequate and acceptable solutions are not found, it can impede progress of the project contemplated. There are examples of nuclear power projects unrealized due to a lack of financing, while in other cases construction has been delayed or cancelled due to unforeseen financing constraints. When there is no money, work stops; delay or cancellation of projects involving large capital investments is very expensive and can have ruinous consequences.

Nuclear power plants are complex, as is the financing. It would be highly gratifying if simple and adequate financing could be found which would be readily applicable and acceptable for nuclear power plants. That is, a solution satisfying the investors/ owners/operators, the governments of the countries concerned, the financing institutions and the suppliers of goods and services.

However, experience shows the financing challenge has effectively been met in the past and can be met now. To date, more than 500 nuclear power reactors have been successfully financed and built. There is no reason to assume that financing will become an impediment to further nuclear power projects where other essential conditions are met, or that nuclear power will cease to be a viable option due to worldwide financing constraints. Recent experience in nuclear projects where construction has started, as well as in projects currently in the acquisition stage, confirms this.

2. SPECIAL FEATURES OF NUCLEAR POWER PLANTS

From the financing point of view, nuclear plants have some special features that should be considered. The principal ones are:

- Large investment;
- Long lead and construction times;
- Complex technology;
- Regulatory risk; and
- Political risk.

Nuclear plants are capital-intensive compared with alternative energy sources. Fossil-fueled electricity generating plants (coal, oil, gas) producing an equivalent amount of electricity, are less expensive to build. Gas-fueled plants in particular, are much less expensive alternatives. Hydroelectric plants also tend to be capital-intensive, except where very favorable site conditions lead to relatively low costs. However, on most such sites, hydroelectric projects already exist, so new projects tend to be more expensive, often costing more than nuclear plant. As to the renewable options (wind, solar, biomass, etc.), costs to produce equivalent amounts of energy are, as a rule, considerably higher than for nuclear plants.

A long time is required for practically all stages of nuclear power project planning and implementation. Relatively long construction times especially have a major impact on overall capital requirements, which must be financed before the plant produces electricity and before there are revenues. There are also risks of delays and cost-overruns, usually perceived as greater for nuclear projects then for fossil-fueled alternatives.

Nuclear technology is complex, a challenge to engineering and management and the supply industry in general. It is an area where there can be no compromise on quality and safety considerations are of paramount importance. New developments in high technology areas constantly take place, but require extensive research, development, and testing before they can be implemented with confidence. And even with all precautions, technological problems may arise with impacts on economics. All this implies technological risks.

To assure nuclear safety to protect the public and plant personnel, nuclear power functions in a highly regulated environment. Nuclear power plants must comply with all rules, regulations, procedures, authorizations and conditions set by national regulatory authorities. This has economic and financial implications. The regulatory environment is not static. It constantly evolves and tends to become more stringent, setting more and more demanding goals and conditions to be met. This constitutes a regulatory risk that can lead to delays in construction, changes, modifications, and corresponding additional costs. This regulatory risk continues after construction is complete, during the whole life of the plant, and can lead to costly backfitting or even premature shutdown of the reactor, with impact on the expected revenues and the finances of the owner/operator.

Politics and nuclear power are inseparable. National and international politics affect nuclear power, and eventual changes from the conditions prevailing when a nuclear project is launched and financing is committed, constitute what might be called a "political risk". International politics may affect the market in nuclear technology, fuel, materials, equipment and components as well as co-operation between countries and can have serious consequences for countries dependent on foreign supply. The influence of national politics on nuclear power presents possibly an even greater risk. Governments do not last forever, and when they change, new governments may have differing views of the nuclear option, and may implement corresponding policies. National politics tend to respond to public perception and media attitude regarding the nuclear issue and these may also change.

In general, the special features of nuclear power present increased risk to the investor/owner/ operator as well as to financing institutions, and they must be compensated by greater benefits to ensure financial viability.

3. EVOLUTIONARY WATER COOLED REACTORS

Evolution implies improvement and the evolution of nuclear power technology is a continuing process. Improvements from lessons learned, from research and development efforts as well as from general technology development, are introduced into new designs and projects. The main goals of evolutionary water-cooled reactors, while maintaining reliance on proven technology, are:

- Cost reduction;
- Higher safety levels;
- Better reliability; and
- Shorter construction times

These common goals are pursued but their simultaneous achievement is difficult. Improvement in one aspect may involve penalties in others and thus, compromises must be found which may effectively limit the extent to which goals can be achieved. Whatever the results, the combined effect of the improvements should have a positive influence on financing.

4. PRECONDITIONS TO FINANCING

Several conditions must be fulfilled so that the challenge of financing a nuclear power project can be met with reasonable expectations of success. The principal preconditions to financing are:

- National policy supporting nuclear power;
- Creditworthiness;
- Economic competitiveness;
- Project feasibility;
- Assurance of adequate revenues;
- Acceptability of risks; and
- No open-ended liabilities.

Establishing national energy policies, including those affecting the electricity sector, is a governmental responsibility. These policies constitute a framework and the ground rules governing the function of the supply market in every country. Some countries prefer a highly regulated environment and predominantly State-owned enterprises; others have or intend to implement a free-market economy with liberalization, deregulation and privatization as goals. It establishes the conditions and ground rules for investment and financing decisions made within the energy and electricity sectors. Taxes, duties, levies, and direct or indirect subsidies are the main tools to create the conditions for implementation of policies and to promote or discourage the use of specific energy sources.

Current national policies regarding nuclear power vary widely. In some countries nuclear power not only continues to be a viable option, but is effectively promoted. On the other extreme, there are countries pursuing anti-nuclear policies. In between, political attitudes and policies vary from acceptance as a viable option, neither to be promoted nor opposed, to leaving it as a "last resort", which hopefully will not be needed. The evolution of attitudes through the years has gradually been changing from predominantly pro-nuclear to more passive or even opposing views. Hopefully, this trend will gradually be reversed.

An essential precondition to financing is a national policy environment supporting nuclear power. In countries with anti-nuclear policies, financing nuclear plants is simply no issue as long as such policies prevail. Financing nuclear power is only a challenge in those countries where nuclear projects are considered a viable option and the intention is to implement it.

An investor/owner/operator requesting a loan from a financing institution to complement his own resources, insufficient to fully finance the project he wishes to implement, must be creditworthy. Creditworthiness of the borrower is assessed by the financing institution. If the prospective borrower is found creditworthy, credit is forthcoming, the challenge of financing can be met, and the project can be implemented. Creditworthiness is therefore an essential precondition to financing but is not a "nuclear" issue. It is determined by the financial soundness of the enterprise concerned, as well as the prevailing conditions of the country.

In addition to fulfilling the two basic preconditions, that is, the existence of a national policy supporting nuclear power and creditworthiness of the investor unable or unwilling to fully finance his project with his own assets (the usual case), several other preconditions should be met.

The nuclear project should be economically competitive with available alternatives. Economic competitiveness should be examined in a wider sense than simply the direct costs of energy produced. It should include external costs such as health and environmental damages arising from construction and operation of the plant, not always considered for other energy alternatives. In general, all costs and benefits should be identified, assessed and included in the comparative economic evaluation.

The feasibility of implementing the project should be demonstrated through site-specific studies. Adequate revenues to be generated by the project must be assured. All risks must be considered and assigned to the respective partners best equipped to manage them. Open-ended liabilities should not remain unattended or left for resolution at some indeterminate future date.

If all preconditions are met, financing can proceed with reasonable expectations for finding solutions. Compliance with the preconditions is the sole responsibility of the investors/owners/operators and their governments. The direct involvement of financing institutions comes afterwards.

5. FACTORS AFFECTING THE VIABILITY OF FINANCING

In addition to compliance with the preconditions to financing, which can go from minimally acceptable to highly satisfactory, there are many factors affecting financing. Like other business undertakings, financing institutions are mainly concerned with the assurance of getting their money back together with expected returns, while minimizing their risks. The expected benefits should be commensurate with the accepted risks. Financing institutions are not really concerned with environmental, strategic, moral or emotional issues, except when they affect risk/benefit assessments.

The overall amount of investment capital required is a major factor affecting financing. For large, multiunit plants, the amount of money required normally exceeds the upper limit that financing institutions are able or willing to provide for a single project. Even the largest financing institutions have limited capacities, which is dispersed among many projects and debtors to reduce overall risk.

In nuclear projects, provenness and licensability have a relatively large impact on risk. Contrary to the attitudes of researchers, scientists or designers, financial institutions as well as electric utilities tend to be conservative. They prefer projects relying on well-proven technology, where licensability has been demonstrated. Obviously, too stringent application of these criteria and attitudes would preclude the achievement of technological advances and the introduction of improvements.

Governmental commitments to relevant international treaties and conventions as well as to stable pro-nuclear policies and the project to be implemented, have an impact on the perception of risks, and in consequence on the viability of financing. There should be reasonable assurance that current commitments will endure unchanged, after the next opinion poll or election.

The availability and adequacy of domestic infrastructure (legal, regulatory, manpower, industrial, institutional) is normally not an issue in countries with operating nuclear plants. But for first nuclear projects, it is a risk factor.

Other factors are the characteristics and regulations of the electricity market, the social and economic system of the country, the domestic participation policy, and public attitudes and perceptions regarding nuclear power. In addition to the current situation, trends and expectations for the foreseeable future should be considered from the point of view of financing.

6. FINANCING REQUIREMENTS AND SOURCES

It may be superfluous to state that financing resources must be provided to cover all expenses incurred in the project, up to the time when it can earn sufficient revenues to maintain itself. To the question "Has everything been included?" the usual answer is: "Yes !" However, following a detailed analysis, the answer is generally modified to: "Yes, except.....". And there is a long list of exceptions representing substantial amounts of money. Adequate provisions must be made to cover contingencies, cost-overruns, and events which cannot be foreseen or avoided, and which may have cost implications.

Expenses incurred may be in domestic currencies alone, or in both domestic and foreign currencies. On the other hand, the project generates revenues exclusively in domestic currencies, except in rare cases where electricity is exported and paid for in foreign currency.

The domestic sources of financing consist of:

- Resources of the investor/owner/operator;
- Bonds issued for the domestic market;
- Bank credits from the domestic capital market;
- Donations and credits from public entities; and
- Government budgets.

To the extent that there are eligible domestic suppliers of goods and services for a nuclear power project, these are used and paid for in domestic currency. This means that for a domestically designed and supplied nuclear power plant, practically all expenses are paid in domestic currency. Even for countries importing a nuclear power plant on a turn-key basis, construction and erection must be performed locally, there are local expenses of the owner, and there is always at least some domestic participation in supplying materials, components and equipment.

For expenses incurred in domestic currency, domestic sources of financing are preferred or often, the only option. For countries with scarce capital and competing investment requirements and choices, a high level of indebtedness, a subsidized electricity market with inadequate tariffs, or dependence on government contributions, domestic financing is a major challenge and may be a constraint. There are examples of long, expensive delays in power projects, nuclear as well as hydroelectric or fossil-fueled, caused by problems of domestic financing.

The foreign sources of financing are:

- Multilateral Development Institutions;
- Export Credit Agencies (ECA);
- Supplier's Credit;
- Bilateral Sources; and
- International Capital Market (commercial loans).

Among the foreign sources of financing, Multilateral Development Institutions are usually mentioned first. They include the World Bank Group, Regional Development Banks and Organizations (African, Asian, Interamerican, European) and other institutions such as the Islamic Development Bank, the Arab, the Saudi or the Kuwait Funds for Development.

For nuclear projects , the most important sources are the Export Credit Agencies, such as the US EXIM Bank, the Export-Import Bank of Japan, the Kreditanstalt fur Wiederaufbau in Germany, the Banque Francaise du Commerce Exterieur (BFCE) of France, the Export Development Corporation in Canada, and similar institutions in other countries that are major industrial exporters. The ECAs are complemented by relevant insurance agencies.

Supplier's Credit and Bilateral financing sources are usually complementary to the ECAs, they may also be intended to promote exports, or to provide tied-aid credits, aid loans and grants. Finally, the International Capital Market or Global Money Market provides commercial loans and is gaining increasing importance.

Contrary to domestic financing, which may constitute a major challenge for nuclear projects, financing is usually readily available to a creditworthy investor through foreign sources, especially those intended to promote exports or those involved with the exporters.

7. TERMS AND CONDITIONS OF FINANCING

The terms and conditions applying to domestic financing are essentially country-specific but financing energy projects, including nuclear plants, differs for State-owned and privately owned enterprises. Where electric utilities are State-owned or under majority control of the State, it is directly or indirectly, the main source of financing. This is the case for nuclear power in most developing countries and in many industrialized countries. The terms and conditions, under which the State contributes to a nuclear project, where needed, are determined by the State, in each specific case.

It is noted that when important State financial contributions come directly from the budget, this might represent a major risk. Budgets are approved for much shorter periods than required to build a nuclear plant, and if the need for budgetary cuts or restrictions appear, which is not unusual, the results might cause delays, suspension of work or even cancellation. For this reason, assured financing through any other domestic source seems preferable, even if terms and conditions appear less favorable.

When nuclear power projects are launched by privately owned utilities, the prevailing general conditions and terms of the domestic money market apply, even though subject to constant change. State contributions are not expected except for demonstration or prototype reactors and the utility makes all necessary financing arrangements. This might be a constraint, because investors would have practically no access to preferential financial treatment from the Public sector. Also, private investors are normally reluctant to launch long-term capital-intensive projects, when short-term low-investment options are available. To overcome this reluctance, incentives are needed, not from commercial banking but from government, where national policies and measures to promote implementation of these projects are important.

For foreign financing, Multilateral Development Institutions generally offer the most favorable terms. They provide "soft" loans, but under "hard" conditions, which might be difficult to meet or accept. Nuclear power projects in developing countries have never been financed by any Multilateral Development Institute, and there is only one example of financing a nuclear project in an industrialized country (World Bank for Latina, in Italy), some forty years ago. In theory the Multilateral Institutions are willing to consider requests for financing nuclear projects but in practice, they avoid it.

The financing terms of aid credits, aid loans or grants are even more favorable than those of Multilateral Institutions but application for nuclear power plants is forbidden by OECD Consensus. The OECD Consensus is an "Arrangement on Guidelines for Officially Supported Export Credits", which includes a "Sector Understanding on Export Credits for Nuclear Power Plants". This consensus was agreed upon "to avoid excessive competition in the terms and conditions of export finance among OECD Member Countries" and is revised from time to time [1]. It stipulates the terms and conditions that may not be underbid by Member Countries and effectively limits competition among them. The terms and conditions for nuclear powers are less favorable than for other energy sources by considerable margin placing a definitive penalty on the nuclear option. But even under these constraints, strictly observed in export credit arrangements, Export Credit Agencies are the preferable financing sources for nuclear projects.

Export credits are limited by the overall export value. In addition, financing local costs and capitalization of interest accruing during the grace period together cannot exceed 15 per cent of the export value and the maximum repayment period cannot exceed 15 years. Interest rates are 75 basis points (0,75 per cent) higher than the applicable rates for non-nuclear projects. In addition to these basic terms covered by the consensus, other terms and conditions apply, which up to a point, can be negotiated. They include fees (commitment, management, insurance, commissions), grace period, modality of repayment, advance payment, etc. Export credit is always restricted to the project itself and tied to the supplier/exporter. Government guarantees are required as well as export credit insurance.

If financing by ECAs is insufficient, it can be complemented by Supplier or Bilateral credits, and by commercial loans in the International Capital Market under conditions and terms normally less favorable. Foreign financing is always in foreign currency, and must be repaid in the same currency, even if used for domestic expenses. Therefore, as a general rule, foreign financing should be restricted to imported supplies.

8. FINANCING APPROACHES

Conventional financing has been used in the past and is currently used for nuclear power plants under construction or in the acquisition stage. This approach should continue to be available to investors, and will apply to evolutionary water-cooled reactors.

The conventional approach consists of multi-source financing, where a complete package is put together covering the entire cost of the project. The first source is the investor/owner/operator responsible for building and operating the plant. His resources constitute the basis of the package, and should cover a sizable portion of the overall investment. In addition, bond issues, domestic bank credits etc. and, in case of State-owned or controlled enterprises, donations and credits from public entities or the governmental budget, should complete the financing for the domestic expenditures. This approach is basically similar to any power project except that for a nuclear project, the perception of risks is higher, and the terms and conditions of the credits, less favorable. A financially sound utility should be able to meet this challenge. It is once again emphasized that firm commitments for domestic financing are essential. Loans in foreign currency could also be used to cover part of the domestic expenditures, but this is not recommended.

When importing from abroad, the conventional approach for the imported portion is to start by inviting financed bids. Export credits should form the basis of the foreign financing package because these have the most favorable terms and conditions among all available sources. Also, the objective of the ECAs is to provide support to the suppliers and potential exporters of their respective countries. The export value of a nuclear power project may be too large to be covered by a single agency. In this case, suppliers from several countries may join in a consortium subdividing the scope of supply and involving several ECAs. The complete financing package may be put together by several ECAs, adding the supplier's credit, bilateral sources and commercial banks operating in the international market. Such multisource financing arrangements are quite normal. The overall terms and conditions may be somewhat less favorable than that from ECAs alone, but they would still be acceptable to the importing utility, and certainly more favorable than that from the international money market alone. With this approach, 100 per cent financing of the foreign portion should be achievable, if required.

Completing conventional financial arrangements takes some time; detailed negotiations might last a year or even more. Comprehensive governmental guarantees are needed. All parties share a common interest in getting the project underway and this helps reach agreements acceptable to all.

There are innovative financing approaches offering attractive alternatives to the conventional approach which could in principle be applied to nuclear projects. Some have already been used to finance fossil-fueled or hydroelectric plants in developing countries.

The BOT (Build, Operate, and Transfer) approach is based on a concession to foreign investors to build, operate and after a defined period, transfer an operating plant to the host government. The BOO (Build, Own, Operate) approach is basically the same, except the plant is not transferred. In both approaches the investor is given guarantees for marketing his product and assurances regarding revenues. These approaches are intended to attract foreign capital in the form of non-government debt for power plants.

Counter-trade or barter arrangements involve exporting local products to the country of the foreign supplier, in exchange for the supplies received or for part of them. It generates foreign currency with local products, which are used to pay off foreign currency debts incurred.

Several other approaches have been envisaged to meet the challenge of financing, such as non-recourse or limited recourse project financing, leasing by local or foreign investors, development of instruments to finance local costs, expanded co-financing operations, revenue bonds with yields tied to enterprise profitability and sale of electricity futures. No doubt, other innovative approaches could be developed and probably will be in the future.

A common feature of all innovative approaches is complex, long and difficult negotiations. There have been attempts in the past to apply some of them to nuclear power projects, such as the BOT, the BOO, or the counter-trade arrangement. These attempts were unsuccessful and were abandoned after failing to reach agreement. This, of course, does not mean that new attempts will be similarly unsuccessful, or that the search for innovative approaches should be abandoned. On the contrary, efforts to find better ways to finance nuclear projects should continue. In the meantime, the Conventional Approach offers a generally satisfactory solution.

9. LESSONS LEARNED - CONCLUSIONS

- More than 500 Nuclear Power Reactors have been successfully financed and built. This has been done in an environment of national policies supportive of nuclear power, and by creditworthy investors.
- While investment decisions are based mainly on economic and financial factors, national policies are based, in addition, on other factors.
- Creditworthiness is not a nuclear issue.
- Financing is a major challenge, but has been met in the past and can be met in the future.
- Current conventional financing approaches are expected to remain viable; innovative approaches may offer better solutions in general, or in special situations.
- The importance of adequate domestic financing must be emphasized. This often constitutes a very real constraint.
- **Evolutionary Reactors with smaller overall investment, shorter construction times, reliance on proven technology, together with predictable regulatory regimes and stable, reliable long-term national policies favorable to nuclear power, should make it easier to meet the challenge of financing nuclear power plants.**

REFERENCE

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SOCIAL-POLITICAL FACTORS INFLUENCING THE EXPANSION OF NUCLEAR POWER

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Abstract

As humanity marches closer to entering a new millennium, it is crucial that we nuclear professionals take renewed stock in the importance of our role in public interactions. The lack of public support has been very influential in bringing the once robust nuclear power construction program in America, Europe and Russia to a grinding halt. In trying to understand the key forces that shape public opinion on technologies such as ours, it becomes clear that the major force is the media enterprise. If we compare the basic driving forces behind successful media with the basic drivers behind successful science, we quickly come to the realization that the media are fundamentally in the entertainment business. Capturing instant attention and holding it are the principal ingredients of success for newspapers, radio, and television. Recognizing that these success drivers are diametrically opposite of those governing good science, we can readily understand why the antinuclear movement has been so successful in orchestrating their message in a package ideally suited to a receptive media enterprise. However, before blaming all our woes on the media, we need to assess at least four technical areas where additional professional work could be of substantial value. These include determining the real health effects of low-level radiation, further developing intrinsic robustness to enhance reactor safety, refining and articulating the environmental ethic associated with the recycling of nuclear waste, and exposing the myth that burying plutonium solves our nuclear non-proliferation problems. We suggest six key ingredients as necessary steps that merit strong consideration in regaining public support for nuclear power. These include 1) seriously addressing the four technical issues summarized above; 2) expressing our key issues and results in language understandable at the high school (or lower) level; 3) continually striving for an open and honest management of the Industry; 4) articulating the BENEFITS of nuclear science and technology as a whole; 5) adopting Decision Analysis techniques wherever possible; and 6) recognizing and taking advantage of appropriate "band-wagons" of public interest issues.

1. INTRODUCTION: PUBLIC OPINION AND ITS IMPACT ON NUCLEAR POWER

As the timeline for crossing into the next millennium marches ever faster in our direction, and we start serious planning for the 21st century, it is crucial that we nuclear professionals take renewed stock in the importance of our role in public interactions. Even modest assumptions on population growth indicate a near doubling of world population by 2050 and at least a doubling of the demand for energy as the developing countries strive to attain standards common today in Europe and America. If this energy growth cannot be met without increases in greenhouse gas emissions, we could be generating an environmental blight of global proportions.

Given the need for our fellow citizens on Planet Earth to make proper decisions regarding the energy sources that will be so desperately needed, it is up to us to help them get over the hurdles that have placed our technology in gridlock. Recent history throughout the world is reminder enough that major corrections within the social-political sector are of crucial importance if nuclear power is to become part of the energy solution on a meaningful global scale.

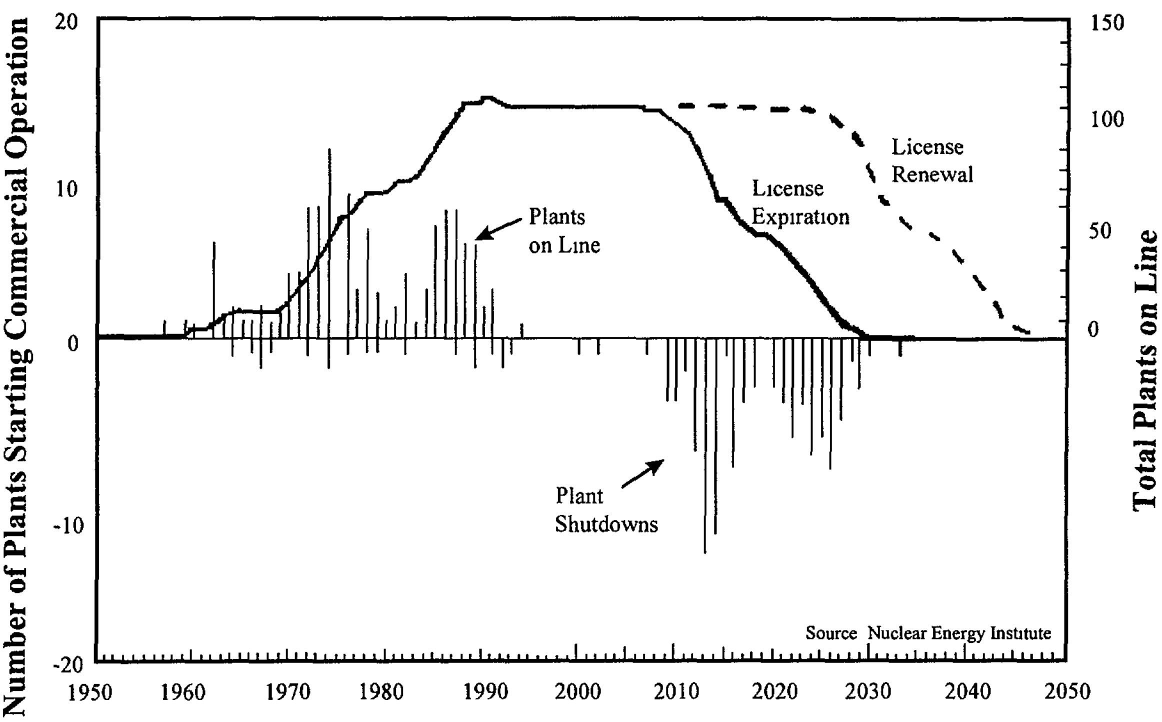


FIG. 1. Starting and Potential Shutdown Dates of Commercial U.S. Nuclear Power Plants

The rise and fall of nuclear power plant construction within the United States is but one illustrative example. As depicted in Figure 1, the first two decades after President Eisenhower's Atoms for Peace program witnessed a flurry of construction activity. Public support was strong, possibly augmented by the pride of marching ahead with home-based technology. But then the years shaped by the Vietnam War and Watergate were entered, and a distrust of big government and big business provided fodder for questioning big, advanced power plants. The Three Mile Island accident in 1979 certainly did not help. As public support waned, new orders stopped. In the fourth decade, some construction continued, but new plant cancellations begin to be announced shortly after the Chernobyl accident. Whereas the diminishing need for new plants certainly had an influence on new plant construction, the correlation between public opinion and the growth of nuclear power is unmistakable.

Similar trends have been experienced in Europe, where massive nuclear power plant construction took place in the 1970s and 1980s followed by a near standstill at present. Again, there was pride in the advances in technology during the early stages. Coupled with the view that "electricity would become too cheap to meter" there was solid public support. But costs have been seen to escalate. Also, the public have come to recognize an attitude within parts of the Industry that can, at best, be described as less than open and honest.

This, together with the growing power of the Green Party (partly fuelled by TMI and Chernobyl), has led to substantial public and political opposition. Decades of hard work in improving public opinion can be written off overnight by incidents such as contaminated transport containers in France and Germany. Currently, in Britain (and even France) there are no new reactor orders, nor do they seem likely within the next decade. In Germany, it is likely that a Red/Green coalition will come into power later in 1998, with a commitment to close several nuclear reactors prematurely. In Sweden, politicians have decided to close the Barseback reactor, despite industry and public opposition.

Many Asian Rim countries were blazing trails with new nuclear power plant construction a decade ago, but here again the pace has slowed considerably. Public street demonstrations in opposition to the latest Taiwan nuclear power plant symbolized rising social opposition to this technology. Both South Korea and Japan continue to build new plants, but the pace has diminished dramatically. A similar story is seen in Russia. Whereas the economic situation there is certainly a major factor contributing to the marked slowdown in new plant construction, growing public distrust of all projects associated with the previous regime is widely evident. Nuclear power particularly suffers from this concern.

All of this is to say that we can no longer focus all our energies on improving the technology of nuclear power and shrug our shoulders at what we perceive to be the futile task of trying to shape public opinion. The consequences of such an approach are far too serious.

2. KEY FORCES SHAPING PUBLIC OPINION

In order to be successful in regaining public support for the next era of commercial nuclear power, we must first understand the key forces that shape public opinion on technologies such as ours.

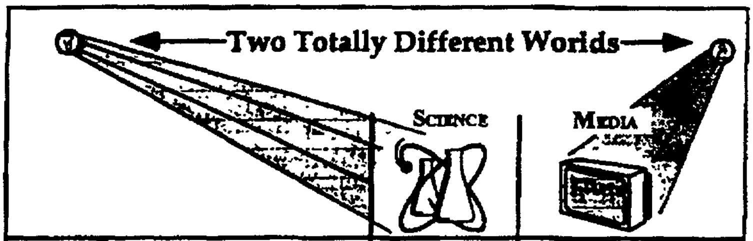
There are undoubtedly several factors involved in forging public impressions of major scientific endeavors, particularly given the plethora of economic and cultural drivers manifest throughout the world. In the Western world, however, there is one force that far out-shadows other influences; namely, the media. The power of the modern mass media in shaping public opinion is no less than awesome. It is so powerful in America that a single point variation in the Nielson rating of a national TV network is worth over $100 million dollars per year in swing revenue. If one were to identify key influences in shaping opinion through a normal life cycle, there is no question that the electronic and printed media completely mask all other forces once a child is old enough to read or soak up images from the video screen.

There is no intrinsic reason why the mass media should be detrimental to the success of an enterprise such as commercial nuclear power. Most technical people would concur that if all the facts surrounding nuclear technology were fairly and accurately reported, the media could be one of the most powerful forces in promoting the development of this awesome humanitarian servant. But fair and accurate reporting does *not* constitute the hallmark of success for a media enterprise that relies on advertising for its livelihood. Faced with the intense pressures of staying in business in a free market atmosphere, the **media are fundamentally in the entertainment business.** Any successful media venture must continually find ways to make its product more appealing than that of its competitors. Careful topic selection, clever packaging, and rapid turnaround are essential ingredients.

Recognizing these elements as essential for staying in business, we can ask how well equipped are the media for dealing with a topic such as nuclear power? Perhaps we should also ask, how well equipped is the nuclear industry for dealing with the media?

Figure 2 summarizes the key ingredients required for success in the worlds of science and the media (1). This comparison, first articulated by Dr. Dixy Lee Ray, former head of the U.S. Atomic Energy Commission and later the governor of Washington State, clearly identifies the immense differences between these two worlds. As noted from this figure, the only common element essential to the success of either endeavor is that they must have a funding source. However, the methods by which such funding is derived lead to vast differences in the mode of operation.

Credibility is the hallmark of good science. Consequently, a good scientist or engineer takes whatever time is necessary to do the work required to arrive at a defensible and well-documented result. This requires an in-depth technical background and a willingness to subject the final product to time-consuming peer review to gain professional acceptance. The media, on the other hand, have very different incentives. Capturing instant attention and holding it are the principal ingredients of success for newspapers, radio, and TV. Whereas credibility may be considered important over the long run, the reality is that there is no time to submit news stories to peers for critique and correction. Deadlines are very real. The crunch of press time is intrinsic to the media business. Further, it isn't practical for any but the largest news organizations to have staff reporters with sufficient training to cover specialized news stories adequately, particularly in the world of science.



	Science	Media
Funding	Program Sponsor	Advertising (or donations)
Principal Success Ingredient	Credibility Of Conclusions	Capturing & Holding Attention
Effect Of Timing	Open-Ended	Instant Turnaround
Reporting Style	Detailed Peer Review	Punchy News Bites
Usual Educational Background	Technical	Broad Non-Science

FIG. 2. Enormous Differences in the Driving Forces for Science and the Media

This is not to say that either science or the media is good or bad. It is simply recognition that the two worlds are miles apart, and it is no wonder that we often get highly distorted media coverage on scientific matters.

Nuclear power likely suffers more than any single enterprise in this regard. Given a hungry press, anxious for stories that are assured to attract attention, what could provide more spice than a "problem" within the nuclear industry? Conversely, what could provide less interest than a nuclear power plant running economically and safely?

Where else can one be guaranteed of attracting an instant audience, anxious to have imaginations stirred by hints of a core meltdown, images of a mushroom cloud, or suggestions of "lethal" releases of radiation—a phenomenon that we cannot see, taste, or smell! These are all makings of prominent and frightening news. It is highly saleable press.

3. TACTICS OF ANTI-NUCLEAR MOVEMENT IN LEVERAGING THE MEDIA

It did not take anti-nuclear activist movements long to recognize the enormous leverage they could garner by crafting their message to feed hungry reporters anxious for headline stories. Elizabeth Whelan, in her classic book Toxic Terror (2), pointed out the four principal techniques employed by anti-nuclear activists to achieve their aims:

(1) Use anecdotal evidence.

(2) Always quote the same handful of "scientists" and liberally cite data largely rejected by the scientific community.

(3) Never mention the advantages of the target substance or process; stress only the negatives.

(4) Time is running out. Take immediate action, whether or not the relevant data are in.

This approach has (unfortunately) been very effective in achieving anti-nuclear goals. In most cases, it forces the Industry onto the defensive. As scientists, we admit that the data are not necessarily 100% complete and in so doing we probably help boost the position of our opponents.

It is refreshing to note that the activists' tactics sometimes backfire. For instance, when Greenpeace orchestrated a campaign to prevent the Shell Corporation from disposing of the Brent Spar oil platform, the media accepted their information without question. When the Industry later showed that their evidence was significantly flawed, it created media hostility to Greenpeace that will, hopefully, create a changed atmosphere in the longer term.

However, that one instance aside, clever packaging, using underlining fear tactics to attract a hungry media looking for bold headlines, has been enormously successful in shaping a negative public opinion toward nuclear power.

4. TECHNOLOGY AREAS MOST RELATED TO PUBLIC OPINION

But before becoming too indignant about an un-level playing field, and blaming the media for all our ills, we need to first take stock on those areas within our technical domain where we may be inadvertently contributing to the problem.

4.1 Health Effects of Low-Level Radiation

Perhaps the singular most important technological issue relating to public opinion is to resolve the question of human health effects of low level radiation. Since it is radiation that the public associates with all fields of nuclear technology, it is only natural that an unfounded fear of radiation could effectively thwart the future of nuclear power (or any other field associated with radiation science) from making its ultimate contribution to the service of humanity.

It is important, therefore, to seriously ponder the way the scientific community has proceeded in assessing the human health effects to ionizing radiation. The current standards, which are based on a linear, no-threshold hypotheses, simply assume that any amount of radiation, no matter how small, is deleterious to the human body. Whereas this is almost assuredly a conservative approach (i.e. it overly exaggerates the potentially harmful effects), there is growing concern among the scientific community that such an approach could be detrimental to societal interests. The reason for this concern is that mounting evidence strongly suggests there is a threshold below which the effects of ionizing radiation to the human body are either completely negligible or even possibly beneficial (3).

Consequently, to insist that radiation at low levels is harmful by definition is causing substantial societal harm. It not only causes the unnecessary expenditure of billions of dollars per year for unwarranted "protection," but it even more seriously instills unfounded fear, thereby seriously threatening the survival of a technology that may be of crucial importance in sustaining life on earth.

It is not reasonable to expect the media, and therefore the public, to minimize or possibly disregard any negative health effects of low-level radiation when influential members of the scientific community themselves cling to the notion that radiation is harmful at any level. Achieving scientific consensus on the real effects of low-level radiation to the human body is, therefore, of utmost importance. There is no guarantee that the public will automatically accept scientific consensus on the matter, but it would certainly constitute a major step toward easing unfounded fears. We can be certain that the public will remain confused as long as the scientists themselves cannot attain agreement.

4.2 Safety

From a safety perspective, the accidents at Three-Mile Island (TMI) and Chernobyl have undoubtedly given the commercial nuclear power industry a negative image. In the case of TMI, despite dominant headlines for a month in leading newspapers around the globe, nobody was killed or even injured. Chernobyl was quite different. Lacking Western safety standards (both from the standpoints of design and operation), this accident was a disaster by nearly any measure. However, we now know that despite the 31 firefighters who died within days of their heroic service, a realistic analysis of the aftermath from this worst-case accident reveals surprisingly diminished long-term effects. Nonetheless, from a public perception standpoint, both accidents left an indelible scar.

It is within this context that the nuclear industry was spurred into designing a new generation of power plants that inherently exhibit substantially higher levels of safety against the possibility of radiation release. Some of these designs have been recently approved by the U.S. Nuclear Regulatory Commission for large-scale production, and the European Pressurized Reactor (EPR) is expected to pass the licensing hurdle in Germany shortly. These represent significant technical steps forward that could be of major importance in easing public fears about "runaway" accidents. Maximizing the safety performance of a plant by building in robust features up front, rather than relying on "fixes" once operational, the technical community is providing a major contribution in the pursuit of gaining public confidence.

However, we cannot afford to commit self-inflicted injuries. The recent issue concerning contamination of transport containers in Europe furthered the impression that the Industry not only fails to control its operations safely, but then attempts to paper over the cracks. This does not work for American Presidents, nor does it work for the Nuclear Industry. Hard work to change our public relations image will only be successful if we control our operations safely and openly.

4.3 Waste Disposition

The question of nuclear radioactive waste disposition currently appears to pose the most significant barrier to the future of nuclear power. It is ironic that this "problem" has gained such momentum since from a scientific point of view it is one of the easiest issues to deal with. The commercial nuclear industry is somewhat unique among industrial enterprises in that it has always sought to concentrate its wastes for disposal, in contrast to other industries that rely on dilution. If one realistically considers the amount of high-level nuclear waste involved on a per capita basis, it becomes immediately obvious that there should be safe ways to deal with this issue.

Whereas it is true that such material remains radioactive for a long period of time, if properly recycled to maximize its energy content, the radioactivity of this material falls below the levels of the uranium from which it was mined within a few hundred years. Emphasizing such a recycling ethic, similar to the recycling ethic for paper, glass, metals, etc. warmly embraced by the environmental community, may eventually be accepted by the public—assuming there is a viable mechanism to get sufficient attention to engage them in an evaluation.

The technologies for successful HLW repositories have existed for years. In fact, HLW repositories could have been built decades ago. Unfortunately, there were not built, and we now find ourselves in the untenable dilemma of being forced to design to essentially impossible standards—standards that have evolved more from a political rather than a technical basis. If such standards were applied to other industries, such victimized entities would also likely be threatened with extinction.

Though of debatable value from the standpoint of scientific need, there are technologies emerging that have the capability of transmuting radionuclides with a long-life into stable elements. This involves 1) advanced chemical partitioning schemes to isolate radioisotopes created in the fission process and 2) neutron bombardment techniques to transmute undesirable radioisotopes into more benign species. There is most certainly an extra expense associated with the implementation of such technologies, but they can be made available if the public insists on bearing this expense.

4.4 Non-Proliferation

Nuclear nonproliferation has been a legitimate concern ever since the development of the first atomic bomb. It is important to recognize, however, that the genie cannot be put back into the bottle. Responsible policy cannot be formulated on the basis of unwarranted fear and half-truths.

As Dr. Glenn Seaborg has repeatedly pointed out, perhaps the most egregious fallacy continually espoused by many of the anti-nuclear groups is that we must bury plutonium to "get rid of the problem." Nothing could be further from the truth. It is well known that the radioactive protective barrier surrounding plutonium in spent nuclear fuel will decay away with time. In little more than one hundred years, the barrier is sufficiently weak that terrorists seeking to acquire plutonium would find such repositories most convenient mines! Emerging analyses associated with long-term energy planning are now beginning to take into account the potential proliferation effects associated with an increasing dependence on plutonium as an energy source. Such studies are confirming that the safest place for the storage of plutonium is inside nuclear reactors (4).There is no intrinsic reason why long-term energy supplies based on utilizing plutonium fuel cannot be safety provided in concert with achieving responsible nonproliferation goals. Technical professionals must be willing to stand up and tell this story.

5. KEY INGREDIENTS TO REGAIN PUBLIC SUPPORT

It would be the height of presumption for us to make dogmatic statements regarding the steps that must be taken to achieve a warm measure of public support for nuclear power. If such a prescription were assured, the nuclear industry would long since have adopted such a course and we would not be discussing these problems today. However, we believe there are necessary steps (perhaps not sufficient) that merit strong consideration.

(1) The technical issues outlined above need to be taken seriously and addressed with renewed vigor. It is difficult to expect the public to feel entirely secure when there is considerably controversy within the scientific community. This is particularly true regarding the health effects of low-level radiation. Given the requirements for a free press, the media will *always* search for controversy. We cannot ignore this reality.

(2) It is simply not reasonable to expect the public (and the media, as their source of information) to understand our technology if we insist on using technical jargon. We *must* find ways to communicate our key issues and results in language understandable at the high school (or lower) level. Continuing to decry a technical illiterate public will not solve our problem. Certainly the quest of tangibly enhancing public appreciation for science by upgrading our school systems is a worthy goal, and it should be pursued with all vigor. But we cannot wait for another generation to make the key decisions necessary for survival in the next millennium. We must prepare our current message in language both understandable and appealing to our current citizenry.

(3) We must at all times strive for an open and honest management of the Industry. Clearly, we want to operate at the highest safety standards, but without the open approach we hand our opponents the silver bullet. Public confidence will never improve if there is any thought that we may have attempted not to keep the public or the authorities fully informed. Under such circumstances we appear, at best, to be dishonest, and at worst, incompetent. Dr. Peter Sandman (5) has repeatedly pointed out the reality of a factor he calls "Outrage." Whereas we technical types normally define risk as the product of probability times consequences, Dr. Sandman defines that product as the hazard factor and then contends that the additional "outrage" factor must be included. Outrage is related to factors such as voluntary (vs. involuntary), control (vs. lack of control), and familiar (vs. unfamiliar). By recognizing such factors are real, it is sometimes possible to deal with them in a constructive manner.

Even if one is not directly in control, it is important that a trust level exist between the two relevant parties. For instance, most airline passengers recognize that they have very little actual control for their safety if the airplane should crash. Yet millions board commercial aircraft every day for at least two reasons. First, they perceive a direct benefit (it gets them to their desired destination), and second they recognize that the pilots are professionals and they want to live just as badly as the passengers do. Hence, the control issue is substantially ameliorated. The nuclear industry would do well to ponder these implications and continually search for ways to build trust.

(4) We must articulate the **benefits** of nuclear science and technology as a whole. It is simply a fact that the tolerance of the public for any risk is exceptionally small if there are no recognized benefits. It is true that the higher the level of education, the higher the tolerance level for the acceptance of risks. But even the most highly educated segments of any public group are reluctant to deliberately expose themselves to a risk unless they can perceive an immediate benefit. The tolerance level declines considerably for those less well educated.

This recognition formed the basis for the new nuclear advocacy group in the United States *called the Eagle Alliance. This movement came into existence when it was recognized that in* 1991 the economic impact of nuclear science and technology in the U.S. was over $300 billion annually, supporting some 4 million jobs. Updated figures for 1995 pushed this total over $400 billion annually. As such, nuclear science and technology in the U.S. represents about 4% of its Gross National Product and some 5% of the total workforce. This aggregate represents an industry larger than the biggest corporation in the nation; namely, General Motors. It is bigger than the entire U.S. airline industry!

The basis for this enormous economic/jobs impact is the myriad of applications already existing for harnessing radioactivity. The benefits of nuclear science and technology are no less than awesome (from insect control to cutting edge medical cures; from developing new food sources to protecting airline passengers from potential explosive devices, etc.). It is essentially impossible for any citizen in a developed country to go through a day without being directly or indirectly served by the marvels of radiation science. The problem is that very few people are aware of these present-day benefits.

Hence, the goal of this organization is to articulate the enormous BENEFITS of nuclear science and technology in everyday life (medicine, food safety, pharmaceuticals, energy, industry, etc.) and to awaken Americans to the importance of further developing this technology to sustain a high quality of life in the next century. If successful in America, this awareness movement could be constructive in the international community as well.

(5) We should give careful consideration to employing Decision Analysis, rather than Risk Analysis. Substantial progress has been made in the technology of risk analysis over recent years, and there is much to be said for the role of this approach to setting priorities. But we must also recognize two key factors that can severely dampen the effectiveness of risk analysis in dealing with public policy. First, as we stated above, there is a strong reluctance for the public to accept *any* risk unless there are clearly perceived benefits. Second, unless we are very careful, defining risk itself can be perceived as a power play. Many people are wary of the ability of statisticians to "set the rules" to make their point.

The Decision Analysis approach, on the other hand, is built around all effected parties coming together and defining up front the issues and the desired outcome. It is structured to achieve win-win solutions. A key element of this approach is to define incentives that can tangibly draw all parties together to achieve mutually desirable goals.

Once success story in the nuclear business is the site cleanup that has been achieved at the West Valley site in New York State. Whereas embarrassingly long stalemates have been encountered in many other cleanup sites in the U.S., one of the key features of the West Valley cleanup pact was to agree that the state would directly pay a portion of the costs. Hence, there was a built-in incentive from all funding parties for the cleanup process to move forward, rather than falling victim to the changing standards and finger pointing all-too-often experienced at other sites.

(6) Finally, in stressing the advantages of nuclear power, we must be prepared to recognize and get on the appropriate "band-wagons" as they come along. This will help move us from the defensive to the offensive.

That is not to say that we should become too opportunistic, because our public are no more foolish than we are ourselves. As we said earlier, unsubstantiated statements like "electricity will be too cheap to meter" eventually rebound against us!

However, we do not appear to be well organized in the current environmental debate. Some nuclear advocates have, of course, already pointed out the enormous environmental benefits of nuclear power—since no greenhouse gases are emitted during normal operations. There are still many scientists who, rightly, point out that the data are incomplete, i.e. while there appears to be a link between global warming and greenhouse gases, the evidence is not yet conclusive. Here we suggest our opponents may have the right tactic; if we wait for the final proof, it may be too late.

Many countries have agreed to the Kyoto protocol for reducing greenhouse gas emissions and it is becoming clear that the increasing world energy demand will not be achievable without an increase in nuclear power generation if we are to meet the Kyoto targets. For example, if the 425 nuclear plants currently operating worldwide were replaced by fossil fuel generation, an additional 2000 million tons of carbon dioxide would be discharged annually.

This is an issue that is becoming more and more a centerpiece of debate as we cross into the new millennium. As such, nuclear power advocates have an ideal climate into which to make their case. It isn't necessary to accept the postulate of a global warming catastrophe. Rather, it is only necessary to recognize that this is a public issue, and as such there is now considerable attention among the public for a scientific debate. We are convinced that the attributes of nuclear power are so strong that an aroused public, actively looking for solutions, will find nuclear power attractive.

6. CONCLUSION

The lack of public support in many parts of the globe for commercial nuclear power is most alarming. As nuclear professionals, we recognize the enormous public good that can be delivered by a properly designed constructed and operated nuclear power industry. The reality is that there is no other power source on the horizon that has the capacity to adequately serve the growing population of the next millennium. Yet unless public support is tangibly improved in the very near future, the nuclear infrastructure could crumble--effectively rendering this technology as an impotent bystander. The humanitarian consequences of such a possibility are almost too tragic to contemplate. Consequently, we technical professionals must find more effective ways to communicate the benefits of our technology to a doubting (or apathetic) public.

Probably the most effective technical tack to take is to insist on the highest integrity in evaluating all relevant data pertaining to the health effects of low-level radiation. Evidence continues to mount that severely questions the validity of the linear hypotheses, no-threshold approach to radiation safety. If scientific consensus can be obtained to clearly demonstrate the effects of a threshold (or even beneficial effects), the current paradigm of viewing radiation as a hazard could change remarkably—possibly ushering in a new era of public acceptance of "things nuclear."

Beyond this, we technical professionals must open the eyes of a slumbering public to the immense benefits of our technology—and do so in the language of the street. We must also recognize the psychology involved with a technology so poorly understood by our contemporary citizenry. As such, we must be willing to turn over control of our systems whenever possible and yield on other matters when we can do so to serve a broader public interest.

Finally, we must be willing to take advantage of public interest in related issues to offer an explanation of the benefits of our technology during times the public is ready to listen. In so doing, we might consider engaging in a Decision Analysis approach, rather than insisting on risk analysis alone. The current concerns about greenhouse gas emissions and global warming provide an excellent springboard to change public perceptions of nuclear power. The Industry could do much more to mobilize its best communicators.

Such steps may not be easy for most of us. But the consequences of failing to try are too severe to contemplate. As professionals, we must be willing to seize the high ground and go beyond our normal comfort zone. We have powerful story to tell, and tell it we must!

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KEY FACTORS IN THE DECISION-MAKING PROCESS

(Session II b)

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REQUIREMENTS AND INTERNATIONAL CO-OPERATION IN NUCLEAR SAFETY FOR EVOLUTIONARY LIGHT WATER REACTORS

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Abstract

The principles of safety are now well known and implemented world-wide, leading to a situation of harmonisation in accordance with the Convention on Nuclear Safety. Future reactors are expected not only to meet current requirements but to go beyond the safety level presently accepted. To this end, technical safety requirements, as defined by the IAEA document Safety Fundamentals, need be duly considered in the design, the risks to workers and population must be decreased, a stable, transparent and objective regulatory process, including an international harmonisation with respect to licensing of new reactors, must be developed, and the issue of public acceptance must be addressed. Well-performing existing installations are seen as a prerequisite for an improved public acceptability; there should be no major accidents, the results from safety performance indicators must be unquestionable, and compliance with internationally harmonised criteria is essential. Economical competitiveness is another factor that influences the acceptability; the costs for constructing the plant, for its operation and maintenance, for the fuel cycle, and for the final decommissioning are of paramount importance. Plant simplification, longer fuel cycles, life extension are appealing options, but safety will have first priority. The IAEA can play an important role in this field, by providing peer reviews by teams of international experts and assistance to Member States on the use of its safety standards.

1. INTRODUCTION

The principles of safety are now quite well known and are implemented worldwide. It leads to a situation where harmonization is being achieved as indicated by the entry into force of the Convention on Nuclear Safety. To go beyond the present nuclear safety levels on existing installations, management of safety and safety culture will be the means for achieving progress. Future reactors such as evolutionary light water reactors have to demonstrate that their safety not only meet the current requirements but in fact go beyond the safety level presently accepted.

To achieve this a number of key factors have to be met:

- technical safety requirements in design ;
- demonstration of decrease in risk to the workers and public and of environmentally friendly operation;
- clear and stable regulatory licensing process;
- gaining public acceptance on the merits of the new design; and
- operating and maintenance costs well contained in order to meet competitive conditions.

But it will not be possible without the prerequisite of having a good safety performance record and no accident on existing installations.

2. TECHNICAL SAFETY REQUIREMENTS IN DESIGN

The IAEA document Safety Fundamentals defines the principles for design as:

1. The design shall ensure that the nuclear installation is suited for reliable, stable and easily manageable operation. The prime goal shall be the prevention of accidents.
2. The design shall include the appropriate application of the defence in depth principle so that there are several levels of protection and multiple barriers to prevent releases of radioactive

materials, and to ensure that failures or combinations of failures that might lead to significant radiological consequences are of very low probability.

3. Technologies incorporated in a design shall be proven or qualified by experience or testing or both.
4. The systematic consideration of the man-machine interface and human factors shall be included in all stages of design and in the associated development of operational requirements.
5. The exposure to radiation of site personnel and releases of radioactive materials to the environment shall be made by design as low as reasonably achievable.
6. A comprehensive safety assessment and independent verification shall be carried out to confirm that the design of the installation will fulfil the safety objectives and requirements, before the operating organization completes its submission to the regulatory body.

They of course all apply to the new evolutionary reactors. In fact the new designs considered have built on some of them such as better prevention of accidents including severe accidents, more robust defence-in depth, increased prevention of human errors, reduction of exposure to radiation of site personnel and reduced releases of radioactive materials to the environment. Demonstration of systematic implementation of defence in-depth is then essential. International reviews can play an important role in this area.

The operating experience of existing installations was essential in developing the new evolutionary reactors which means including from the design stage a number of improvements leading to less demands to be put on the operators and easing the operational procedures.

2.1 Decrease in risk to workers and population

A well designed and tested containment should decrease the frequency of large radioactive releases to negligible levels. This needs to be fully demonstrated both based on deterministic and best estimate probabilistic analysis as well as through defence in-depth.

On site and off site protection to workers and to the population in general should be clearly elaborated through the design features and in the frame of emergency plans and environmental impact assessments.

2.2 Regulatory process

Of utmost importance is a stable regulatory system. This requires an efficient, independent and technically competent regulatory body and a well established safety approach which ensures harmonization in the safety decision making process.

The licensing process needs to be transparent and objective. Predictability and stability of judgement are important aspects to limit the total duration of the process to no more than some five years. The interface between the regulatory body and its licensees should also provide the means for the required quick responses from both sides.

A well established self assessment process leading to efficiently needs to be in place at the regulatory body to ensure that all aspects of nuclear safety, technical and managerial, are being properly addressed. Periodic international peer reviews are an appropriate instrument to provide upper management with an independent assessment and a comparative perspective to similar work going on worldwide. Harmonization of regulatory decisions concerning licensability of the new reactors would also be desirable for increasing public understanding of nuclear safety.

2.3 Gaining public acceptance

Gaining public confidence will require the nuclear industry to perform well on existing installations. The first implies no accidents, the latter involves unquestionable results from safety performance indicators, a recognized international harmonization of nuclear safety, and compliance with legally binding international instruments such as the Convention on Nuclear Safety.

2.4 Operator and maintenance costs

In the competitive environment and to face deregulation, longer fuel cycles and life extension are most appealing but with safety being an overriding priority. Periodic safety reassessment, risk informed decisions and modern I&C to support human factors requirements are essential.

In this scenario, the Agency can play a most important role. This involves provision of a wide range of safety systems based on peer reviews by teams of international experts and assistance to Member States on the utilization of IAEA's safety standards which are to be fully revised by the year 2000.

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PREPARING FOR THE FUTURE BY IMPROVING THE PERFORMANCE OF TODAY'S NUCLEAR STATIONS: THE WANO PERSPECTIVE

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Abstract

The World Association of Nuclear Operators, (WANO) was established in 1989 in the aftermath of the Chernobyl accident with the mission of maximizing the safety and reliability of nuclear power plants by exchanging information and encouraging communication, comparison and emulation among its members. All nuclear power stations in the world are WANO members. WANO conducted an Internal Review in 1997 and its report issued this January confirms that the WANO mission is still valid. As a result of the Internal Review, WANO is taking actions to further improve its programmes. WANO's effort to keep members conscious of safety culture in their daily work at plants is a key element for improving operational safety. WANO will be able to contribute to the future of the nuclear industry by encouraging members to actively participate in WANO programmes which are aimed at improving nuclear safety and plant performance.

1. INTRODUCTION

The World Association of Nuclear Operators, which we call WANO for short, was established in 1989 in the aftermath of the Chernobyl accident with the mission of maximizing the safety and reliability of nuclear power plants by exchanging information and encouraging communication, comparison and emulation among its members.

WANO activities are conducted through Regional Centres located in Atlanta, Moscow, Paris and Tokyo under the integrated leadership of the WANO Governing Board consisting of representatives of these Regional Centres. In addition, WANO has a Co-ordinating Centre in London which coordinates regional activities to enhance co-operation and to enhance effectiveness by enabling close communication and avoiding duplication.

Currently all 440 nuclear power plants in commercial operation around the world and the 130 operators of these plants belong to WANO and work together to fulfil its mission. WANO Tokyo Centre consists of six Ordinary Members listed below running 31 nuclear power stations with 83 plants in operation and 18 plants under construction.

- Pakistan Atomic Energy Commission
- China National Nuclear Corporation
- Nuclear Power Corporation of India
- Taiwan Power Company
- Korea Electric Power Corporation
- Japanese Nuclear Operators

2. INTERNAL REVIEW

Last May, at the 1997 WANO Biennial General Meeting in Prague, the implementation of an Internal Review was announced as a joint idea of Mr Rémy Carle, outgoing WANO Chairman, and Dr Zack Pate, new WANO Chairman, to review the current status of WANO and to guide WANO's future development. At that time, eight years had passed since WANO was inaugurated in 1989. Since its formation, WANO has made Nuclear Network available as a handy daily tool of communication for members and Centres, and it has established various technical programmes to assist in the improvement of plant performance such as Event Reporting, Exchange Visits, Workshops/Seminars, Performance Indicators, Good Practices and Peer Reviews.

Mr Bob Franklin and Mr Ray Hall were appointed to lead the Internal Review. They vigorously visited members all over the world and ended up with 190 interviews with individuals and groups of WANO members. The report summarizing the inputs collected widely from the members was issued this January.

The most essential indication of the Internal Review Report was the confirmation that the WANO mission established at the time of the WANO inauguration is still valid. The report also pointed out that more effort should be made to attract the attention of plant managers so that WANO programmes penetrate into member power plants more deeply.

The contribution of the Institute of Nuclear Power Operations, INPO for short, to serve as a model for many WANO programmes is well known. The Internal Review report indicated that it was a unanimous opinion among WANO members that INPO programmes and INPO's approach to solving problems were so good that WANO should try to adopt INPO methods wherever appropriate.

The report also outlined various areas to be further improved in reinforcing individual programmes. Consequently, programmes were prioritised to increase effectiveness, existing WANO programmes were realigned, and a few new programmes were added. As shown in the new programme alignment, WANO has four programme areas under which some individual programmes are classified.

The new WANO programme realignment is as follows:

- Operating Experience
- Peer Review
- Professional & Technical Development
    - Workshops/Seminars/Courses
- Technical Support & Exchange
    - Good Practices
    - Operator Exchanges
    - Performance Indicators
    - Technical Support Missions

3. WANO PROGRAMMES

Operating Experience, at the top of the list, is deemed the most basic of the WANO programmes. WANO intends to improve this programme and make WANO event reports of such high quality that they cause plant managers to take prompt action to ensure that similar events are prevented at their stations. To increase the effectiveness of this programme, WANO has revised reporting criteria, adopted the IAEA's IRS (Incident Reporting System) coding system, and established a Central WANO Operating Experience team.

Peer review is regarded as the strongest programme in WANO containing elements of all WANO programmes. A peer review is an on-site review using peer knowledge and credibility to offer valuable information to a host plant. The WANO Policy Guideline on Peer Reviews stipulates that a peer review should be a voluntary programme initiated at the request of a WANO member utility, that the scope of a peer review should be decided by the host utility, and that a formal report documenting key issues should be written. Follow-up on the areas for improvement identified is totally at the discretion of a host plant. A peer review is to be conducted for all or some of the nine areas:

- Organization and Administration
- Operations
- Maintenance
- Engineering Support
- Training and Qualification
- Radiological Protection
- Chemistry
- Operating Experience
- Emergency Preparedness

A peer review focuses on how plant people perform their daily work rather than how well plant programmes are written. The team consisting of international peers develops "Strengths" and "Areas for Improvement" referring to WANO Performance Objectives and Criteria as the standard of excellence. Strengths identified by the team may be useful to other member utilities. Areas for Improvement identify where operational improvements are possible to achieve excellence at a host plant. WANO can also assist a utility to develop and implement an action plan to address the Areas for Improvement identified by the review team. In addition, reviewers bring back experience gained during a peer review to be utilized for self-assessment at their own plants.

Professional & Technical Development includes workshops, seminars and various courses. They aim to exchange specific experiences among members in more depth. Workshops/seminars is one of the initial WANO programmes and has been appreciated as an effective programme. Courses are rather new and are designed to reinforce WANO members' areas of weakness. Some of these are INPO courses made available through the WANO channel.

Technical Support & Exchange includes four programmes, namely Good Practices, Operator Exchanges, Performance Indicators and Technical Support Missions. Good Practices are collected and disseminated to be shared widely among members, which enable members to learn from each other's best practices and improve their own operational safety and reliability. Members can search Good Practices when they seek specific, tried and proven methods and ideas for improving performance. WANO focuses on quality when identifying Good Practices, and selected Good Practices are posted on the WANO Web site while all the Good Practices presented by members are accumulated in a database for retrieval.

Operator Exchanges enable members to directly share plant operating experiences and ideas for improvement through face-to-face contact between nuclear power plant staff. Members can share best methodologies and high standards as a means of promoting improvements in nuclear safety and reliability. If things go well and interests and benefits agree, this programme may develop to be a twinning agreement for a longer and more formal phase of information exchange.

Performance Indicators support the exchange of operating experience information by collecting, trending and disseminating nuclear power plant performance data in the following ten key areas:

- Unplanned Automatic Scrams per 7000 Hours Critical

- Safety System Performance
- Unplanned Capability Loss Factor
- Unit Capability Factor
- Thermal Performance
- Fuel Reliability
- Collective Radiation Exposure
- Volume of Low Level Solid Radioactive Waste
- Chemistry Index
- Industrial Safety Loss-time Accident Rate

WANO Performance Indicators provide a common standard and a quantitative indication of plant performance for self-assessment and comparison with other plants for improvement. WANO members share plant-specific data to allow consistent comparison of performance and encourage emulation among member plants, which can be utilized as a management tool.

Technical Support Missions is a new programme intended to establish within WANO the capability to provide technical service to meet members' specific needs and requests. WANO wishes to respond to members' needs within the limit of WANO's resources, ability and expertise.

To facilitate and activate these WANO programmes, WANO has its own secure network called WANO Network which is used as a tool for direct contact among members and Centres.

4. UNIQUE FEATURES OF WANO

WANO has some unique features. First of all, as explained in the earlier section on WANO membership, all nuclear power plants in the world are WANO members. This means that through WANO, members can learn from the experience of others, whether it is good or bad -- which is a great advantage.

Secondly WANO members have a common aim, that is to improve operational safety in their power plants. Plant managers without exception hope that safety culture roots deeply in the minds of plant staff in their daily work, and that the whole plant makes every effort to carry out their jobs with safety culture in mind. However, safety culture is very vulnerable. No matter how hard work plant people work, once they think their plant is safe, the safety of the plant starts decaying in that instant. Let me remind you that safety culture is a daily thing, and can never be completed. Modesty is essential for the people at a plant because self-satisfaction immediately destroys safety culture. Participation in WANO programmes and communicating with various fellow plant staff all over the world can keep plant people alert and encourage unfailing efforts for safety. And this is what WANO can be proud of.

Another feature of WANO is that it is a private organization where voluntary participation and mutual cooperation of members are essential. Since the nuclear industry is so interdependent, members must assist a plant with problems without expecting any reward. If something serious happens at any plant in the world, no other plant can avoid its influence. We say in WANO that we are only as strong as our weakest plant. When a weaker member becomes stronger, the entire nuclear industry becomes stronger and that is WANO's aim -- improved safety and reliability of nuclear power plant operation.

5. RELATIONSHIP WITH THE IAEA

While the roles of the IAEA and WANO are different, they are complementary when it comes to the nuclear safety of power generation. WANO aims to maximize the safety and reliability of nuclear power plant operation while IAEA covers wider range of activities. As Dr Blix, former Secretary General of the IAEA, stated on the occasion of 1995 WANO Biennial General Meeting in Paris, "IAEA should not do what WANO can do." Therefore, the organizations co-ordinate their activities to prevent overlap. Meeting safety regulations is the minimum requirement for a plant. Additional spontaneous efforts of operators are essential to achieve safety in the true sense of the word. WANO can fulfil its mission simply by implementing its activities.

6. CONCLUSION

As the title of this presentation suggests, "Preparing for the Future by Improving the Performance of Today's Nuclear Stations," the future of the nuclear business depends on the enhancement of the performance of nuclear power plants. Nuclear generation today faces difficulty in many countries, and the enhancement of performance of nuclear power plants will be a substantial factor for the future success of the nuclear business. In this context, WANO can contribute to the future of the nuclear industry by providing a forum through which nuclear utilities world-wide can improve their safety and reliability.

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TECHNICAL AND INSTITUTIONAL PREPAREDNESS FOR INTRODUCTION OF EVOLUTIONARY WATER COOLED REACTORS

XA0053544

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Abstract

Since 1982, U.S. utilities have been leading an industry-wide effort to establish a technical foundation for the design of the next generation of light water reactors in the United States: the Advanced Light Water Reactor (ALWR) Program. This program provided a foundation for a comprehensive initiative for revitalizing Nuclear Power in the U.S., as set forth in the Nuclear Energy Industry's "Strategic Plan for Building New Nuclear Power Plants." The Strategic Plan contains fourteen building blocks, each of which is considered essential to building new nuclear plants. At its inception, the ALWR Program envisioned new plant orders in the US shortly after the turn of the century, and geared its milestones and deliverables to enabling ALWRs as an option for utilities by about 2000. However, in the U.S., new orders for nuclear plants are not imminent. There are three primary reasons for this - the lack of demand today for major new construction of baseload capacity, the economic and structural uncertainty associated with deregulation, and the lack of an assured resolution to public concerns over long term management of nuclear waste. Deregulation will likely drive further consolidation of the electricity business, as evidenced in recent nuclear utility mergers, acquisitions, and plant purchases by the larger utilities intent on remaining in the generation business. Deregulation has focused attention on some of the inefficiencies in the current regulatory regime for nuclear energy, and is likely to drive the U.S. government to find more efficient and less expensive ways of providing adequate protection of public health and safety. Deregulation has also focused the industry on the significant variations in production costs among plants, fueling the belief that the industry as a whole can make further improvements in this area to match the stable, low cost performance of the top ten plants. Finally, deregulation has focused the nuclear industry on the imperative for ensuring that total busbar costs for ALWRs are competitive with non-nuclear options.

1. INTRODUCTION

Since 1982, U.S. utilities have been leading an industry-wide effort to establish a technical foundation for the design of the next generation of light water reactors in the United States. Since 1985, the utility initiative has been conducted for the utilities through a major technical program managed by the Electric Power Research Institute (EPRI) -- the U.S. Advanced Light Water Reactor (ALWR) Program. In addition to the U.S. utility leadership and sponsorship, the ALWR Program also has the participation and sponsorship of the U.S. Department of Energy (DOE), international utilities, and NSSS vendors. International utilities include KEPCo (Korea), Taipower (ROC), ENEL (Italy), GKN (The Netherlands), JAPC (representing the Japanese utilities), EdF (France), UNESA (representing the Spanish utilities), VDEW (representing the German utilities), UAK (representing the Swiss utilities), Tractabel (Belgium) and British Energy. Participating NSSS vendors include ABB-CE, General Electric, and Westinghouse.

The main goal of the ALWR Program has been to develop a comprehensive set of design requirements for the ALWR, and to use those requirements as the technical basis for achieving standardization and regulatory stability in new plant design, construction and operation. The ALWR Utility Requirements Document defines the technical basis for improved future LWR designs, and has become an international standard that has been used for bid specifications for new plant orders. The program has sponsored extensive design development, testing, safety and cost analysis, construction planning, and regulatory review for four new ALWR plant designs.

The ALWR program provided a foundation for a comprehensive initiative for revitalizing Nuclear Power in the U.S., as set forth in the Nuclear Energy Industry's "Strategic Plan for Building New Nuclear Power Plants", published in November 1990 and updated annually. Before 1994, this initiative and its Strategic Plan were coordinated by the Nuclear Power Oversight Committee (NPOC); after 1994, they were coordinated by the Nuclear Energy Institute (NEI). The Final Report of the Strategic Plan was issued in May 1998. The Strategic Plan contains fourteen building blocks, each of which is considered essential to building new nuclear plants. EPRI's responsibilities under the Strategic Plan were the Utility Requirements Document, the First-of-a-Kind Engineering (FOAKE) program, Siting, and support to the Advanced Reactor Corporation for other project-specific building blocks. NEI's responsibilities under the Plan included regulatory stabilization and all the institutional building blocks. The Institute of Nuclear Power Operations (INPO) was responsible for life-cycle standardization.

The next nuclear plants ordered in the U. S. will be Advanced Light Water Reactors (ALWRs). Two types have been developed: large units (about 1300 MWe) called "evolutionary" ALWRs and mid-size units (about 600 MWe) called "passive" ALWRs. The term "passive" refers to its safety features that depend on natural processes such as gravity and buoyancy, in contrast to powered equipment such as pumps. Based on the IAEA convention for the term "evolutionary designs" (as contrasted to "revolutionary designs"), the U.S. considers the mid-size passive ALWR to be an evolutionary plant in the context of this symposium, since it is based on proven technology and will not require a prototype.

The conclusion that new nuclear plants in the U.S. will be ALWRs is based on many factors:

- the extensive operating experience with today's light water reactors (LWRs);
- the extensive infrastructure and regulatory basis that has been established for LWRs;
- the major reliance on nuclear energy today (~100 GWe of U.S. capacity -- ~20% of total);
- the imposing challenge of duplicating this infrastructure and regulatory basis for non-LWRs;
- the major improvements over LWRs achieved in ALWRs (safety, simplicity, maintainability);
- the known, favorable economics of ALWRs compared to non-ALWR nuclear options;
- the achievement of Design Certification by NRC for the ABWR and System 80+ in 1997;
- the confidence that the AP600 mid-size passive design will achieve Certification after having received its Final Design Approval in 1998.

At its inception, the ALWR Program envisioned new plant orders in the U.S. shortly after the turn of the century, and geared its milestones and deliverables to enabling ALWRs as an option for utilities by about 2000. The project-specific milestones for this objective have largely been met, and remaining milestones will be met before or shortly after the turn of the century. What was not anticipated at the inception of the ALWR program was the intractability of some of the institutional challenges to new plant construction in the U.S. The dominant challenge today to new construction of any large base-loaded power generating station in the U.S. is the economic deregulation of the power generation sector of the electricity business. This deregulation process will take many years, and is sufficiently unpredictable at this point that most decisions on new baseload generation are being deferred. Where essential, lower capital cost capacity additions (e.g., gas-fired combustion turbines) are being made, even if their fuel costs are higher.

As the U.S. looks to the first decade or two of the next century, a very promising picture emerges for expanded reliance on nuclear energy:

- Most current plants continue to operate economically, with all objective indicators of plant safety and performance continuing to improve, a trend that is likely to continue;
- Increasing appreciation among energy policy and political leaders that nuclear energy is an essential part of a balanced energy supply and environmental protection strategy for the next century – a strategy that exploits the low life-cycle cost of nuclear and its clean air benefits;
- The costs of nuclear generation are likely to remain constant or continue the historical decline of the last decade, due to continued efforts to improve efficiency, increase capacity factors, reduce outage times, control O&M costs, etc. By comparison, fossil generation may face increased capital and operating costs associated with increased environmental controls;
- Positive trends in public acceptance of the need for nuclear energy as part of the energy mix;
- Likely progress on reducing regulatory uncertainty and barriers to new construction;
- Likely progress on spent fuel storage and disposal issues.

However, in the U.S., new orders for nuclear plants are not imminent. There are three primary reasons for this – the lack of demand today for major new construction of baseload capacity, the economic and structural uncertainty associated with deregulation and the lack of an assured resolution to public concerns over long term management of nuclear waste. These issues are inter-related and will take years to resolve. Moreover, the life extension of current plants, via U.S. regulations that enable a 20-year license renewal, is the more urgent issue, from both an industry perspective and a national energy strategy perspective. The fact that most utilities will address the license renewal of their current plants first, before building new ones, has other benefits as well, such as providing a clearer picture of the regulatory environment for new plants.

Deregulation will likely drive further consolidation of the electricity business, as evidenced in recent nuclear utility mergers, acquisitions, and plant purchases by the larger utilities intent on remaining in the generation business. Deregulation has focused attention on some of the inefficiencies in the current regulatory regime for nuclear energy, and is likely to drive the U.S. government to find more effective ways of providing adequate protection of public health and safety. Deregulation has also focused the industry on the significant variations in production costs among plants, fueling the belief that the industry as a whole can make further improvements in this area to match the stable, low cost performance of the top ten plants. These lowest cost plants are also high performance plants from a safety and regulatory perspective.

Finally, deregulation has focused the nuclear industry on the imperative for ensuring that total busbar costs for ALWRs are competitive with non-nuclear options. Industry can anticipate some increases in future fossil generation costs from environmental regulations, but cannot rely on such potential future cost increases as a panacea for off-setting the high up-front capital costs for nuclear energy. Industry in the U.S. is looking at the possibility of further cost-saving enhancements to ALWR designs and construction methods, as well as life cycle cost savings from standardized practices, regulatory efficiencies, and shared infrastructure support (e.g., common training, engineering, maintenance facilities and personnel).

2. HISTORY OF THE UTILITY ALWR PROGRAM

U.S. utility executives that shared a vision for a new nuclear era exercised this leadership in the early 1980s, and established, under EPRI, an ALWR program to develop detailed functional design requirements for all future ALWRs, and to help them facilitate a utility-driven framework for future nuclear plants. These utility executives, after having accumulated three decades of nuclear operating experience, saw a need to exercise direct leadership over future nuclear plant designs and to improve the operational and regulatory frameworks they would come under. Utilities carry the ultimate responsibility for the safe and economic operation of their nuclear plants. Utilities were achieving improved performance in currently operating plants, and saw the need and opportunity for ALWRs that would allow for even better cost and safety performance.

EPRI was asked in 1983 to conduct surveys and assess the prerequisites to reopen the nuclear option. EPRI surveyed a broad cross-section of U.S. utility executives in 1983-4 and found that a number of serious institutional hurdles stood in the way of resuming nuclear construction in the U.S. The U.S. licensing process was unworkable, hundreds of regulatory issues and concerns had been identified but not resolved, state and local interference and rate control measures were draining resources from construction projects, no plan was in sight for resolving the nuclear waste issue, and public opinion was vacillating. These findings were revisited formally in 1990 in formulating the NPOC Strategic Plan.

The utility executives also concluded that industry could not address institutional and design issues in series -- we needed to make sure that when the institutional barriers were lowered, promising designs were available. Utilities made it clear that new ALWR designs must be both cost-competitive and capable of meeting all regulatory requirements in clear and demonstrable ways. Utilities needed plants that were safer, simpler and less expensive to construct, operate and maintain, plants that would have very high availability, and plant designs that reduced the potential for operator error. Utilities have learned from experience that proper planning and investment up front can pay large dividends in reduced operations and maintenance (O&M) costs over the life of the plant. This focus on quality, protection of plant investment, and reduced O&M burden all contributed to increased public safety. Finally, essential standardization would fail to materialize if utilities, representing the customers of future plants, didn't agree on the design and operational choices being made, and insist they be applied consistently.

Utilities also saw the need and benefit of broad international utility participation, so design requirements would reflect worldwide experience and the spectrum of future needs. Utility-driven requirements should serve as customer specifications worldwide, and should resolve open regulatory issues so that designs would invoke high confidence of approval by the regulators.

The design concepts available in the mid 1980s needed help from both industry and government to achieve success. Utility assistance was needed to ensure comprehensive customer input to the design. Both government and utility industry financial assistance was needed to bring these designs to a much higher level of design completion and standardization than had been past U.S. practice. Utilities established a requirement that about 90% of the engineering must be complete before construction would begin on a new plant. Government assistance was forthcoming, because both Congress and the DOE saw the public benefits of greater standardization and engineering completion. DOE assistance and expertise would be critical to success.

2.1 Phase 1: Program Planning and Development; Regulatory Stabilization

The 1983-4 EPRI survey of utility executives showed broad support for safer, simpler plants with greater design margins -- a fundamental change from the historical tendency to improve safety via increased design complexity. Utilities also advocated reliance on proven technology as an essential policy for future designs. Utilities supported a continuation of LWR technology, and insisted that radical departures from proven design features would not be welcomed, especially if full prototype demonstrations became necessary. Another valuable insight from the utility survey was the need to evaluate the feasibility of smaller nuclear plants -- in the 600 MWe range -- for use by smaller utilities or utilities with slower load growth. Utilities wanted to retain the option for evolutionary designs, but were particularly interested in smaller, simpler designs.

The key objective of Phase 1 was to develop a basis for regulatory stabilization. Over 700 issues had been identified by the Nuclear Regulatory Commission (NRC) as open regulatory issues for future designs. ALWR Program objectives included achieving high assurance of licensability by establishing design requirements that would resolve the open licensing issues. A process was created in cooperation with the NRC to categorize the open issues and to identify those that needed priority attention. The list of 700 issues was then reduced to roughly 50-70 open issues.

2.2 Phase 2: The ALWR Utility Requirements Document

The utility executive surveys conducted in Phase 1 revealed strong support for a utility-driven process of compiling user requirements for advanced reactor designs. What was desired was a common, standardized set of specifications applicable to both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs) that utilities could agree to, and that the reactor designers and NRC could accept. In response, the ALWR Program produced a set of detailed comprehensive technical design requirements for ALWRs. EPRI's Board of Directors approved $20 million for their development, but before the effort was completed, many international utility participants joined the program, providing additional leadership, perspective, and resources. The *Utility Requirements Document (URD)* was started in 1985 and completed in 1990. The URD is a living document that has undergone many revisions since first issuance in 1990. The URD covers the entire plant up to the grid interface, forming the basis for an integrated plant design, i.e., nuclear steam supply system and balance of plant. It emphasizes those areas which operating experience has shown are most important to the objective of achieving an ALWR which is excellent with respect to safety, performance, constructibility, and economics.

Since the ALWR Utility Requirements Document was to be -- by definition -- a utility consensus document, a strong Utility Steering Committee (USC) was created to guide the program and to make the tough technical decisions. The USC of the 1980s consisted of over 20 senior utility executives from U.S. and international utilities who were directly involved in every phase of the program work, and who reviewed and approved all program deliverables.

Initially, the URD was defined as a set of 13 separate chapters, each specifying utility requirements for a unique functional area of Evolutionary ALWR design, such as Reactors Systems, Building Design and Arrangement, and Man-Machine Interface Systems. It was submitted to NRC for review between 1986 and 1989. This set of utility requirements for the Evolutionary Plant later became Volume II of the overall Requirements Document Structure.

As the feasibility studies for the smaller ALWR plant option progressed, it became clear that the concept of a smaller, simpler plant with passive safety features had great potential, and a decision was made to develop a parallel set of requirements for the Passive Plant. The Passive Plant requirements were designated Volume III. Volume I: "ALWR Policy and Summary of Top-Tier Requirements" was published in April 1990. Volumes II (Revision 1) and III (initial version) were completed and submitted to NRC in September 1990. NRC issued its final SER on Volume II of the URD in August 1992, and issued its final SER on Volume III in August 1994.

As the URD was developed, various ALWR design teams in the U.S. interacted extensively with the USC to make sure that these design requirements could be implemented. The four designs submitted to NRC for review included:

- Evolutionary ALWRs (1200 to 1350 MWe):

  -- General Electric's Advanced Boiling Water Reactor (ABWR)

  -- ABB-Combustion Engineering's System 80+

- Passive ALWRs (about 600 MWe):
  - -- Westinghouse Electric's AP600
  - -- General Electric's Simplified Boiling Water Reactor (SBWR) [later deferred]

2.3 Phase 3: Detailed Design Development of ALWR Passive Plants

The third phase of the ALWR program began in 1989. It involved utility support and leadership in the design and design review of the two specific Passive Plant designs selected for further development by the DOE. Phase 3 addressed the design engineering needed to achieve NRC Design Certification.

The U.S. DOE signed major contracts with both Westinghouse and General Electric in 1989 for the detailed design development and certification of the AP600 and the SBWR. DOE committed $100 million to these efforts. EPRI supported these two projects by providing a significant portion of the required financial support, as well as an extremely vital program of utility participation in an independent, in-depth technical review of each contractor design. In-depth design reviews verified that these passive plant designs conformed to the Passive Plant URD. As with Phase 2, extensive international utility support developed for this effort.

Phase 3 also involved significant interaction with the NRC, as generic issues with unique passive plant solutions were resolved. In many key areas, the utilities had specified requirements more stringent than current regulations, for the sole purpose of providing "margin to the regulations", thereby providing operating flexibility and greater assurance of licensability. Current regulations were interpreted for passive safety grade systems, and critical issues were resolved, such as how regulations should treat the active non-safety systems in the passive designs.

2.4 Phase 4: First-of-a-Kind Engineering

The next stage of design development, called "First-of-a-Kind Engineering" (FOAKE), covered the non-recurring engineering outside the scope of the NRC Design Certification. It was funded by the DOE, the nuclear utilities through EPRI, and the reactor design teams selected in a competitive process. The FOAKE goals were:

(1) Complete engineering on certified designs in sufficient detail to define firm cost and schedule estimates and prepare for construction of standardized ALWR plants.

(2) Ensure that an institutional infrastructure is in place to provide resources and manage completion of detailed design.

(3) Define the process to achieve commercial standardization, that is the design standardization beyond that required for certification.

The DOE funding plan for FOAKE called for $100 million over five years on the condition that the private sector would match that funding. Congress appropriated funds starting in FY1992. Multi-year authorization for FOAKE was a key provision of the Energy Policy Act of 1992.

NPOC designated the existing Advanced Reactor Corporation (ARC) as the organization responsible for raising the utility funds necessary to provide up to half of the private sector contribution, with the design teams to be selected for FOAKE to contribute the other half plus any additional resources needed to meet design objectives. Sixteen utilities pledged to contribute, and sufficient funds were identified to meet the utility funding target of $50 million. The design teams selected ultimately contributed in excess of $100 million.

A Cooperative Agreement between ARC and DOE was signed in February 1992 that gave the utilities funding the program the responsibility to carry out the selection process, as well as significant rights to make the technical decisions to be reflected in future designs. An ARC-EPRI Agreement was signed that same month for administration and technical management of the program. An RFP was issued for design selection, and procedures for the selection process were established. It was ARC's intent that at least one evolutionary and one passive design be selected initially for FOAKE funding. ARC announced their selection of the ABWR and the AP600 designs in January 1993.

The ARC Board of Directors was comprised of the Chief Executive Officers (CEOs) of the 16 contributing utilities. They have established a Utility Management Board, a Utility Sponsor Group for each design, and an Executive Director and staff to oversee the work.

2.4.1 Phase 4 to Present

Progress from 1990 to date on enabling new plant orders in the U.S. has been governed by the Nuclear Energy Industry's Strategic Plan for Building New Nuclear Power Plants, discussed below. This Plan provided a stable vision and accountability for achieving the goals, despite significant organizational changes in industry. Efforts in the 1980s, prior to publication of the first Strategic Plan, focused on licensing reform and nuclear waste issues. The task of resolving nuclear waste issues was assigned to the Edison Electric Institute. Efforts to address licensing reform, via increased standardization and regulatory approval of designs before construction began, were assigned to the Atomic Industrial Forum (AIF), later transferred to the Nuclear Management and Resources Council (NUMARC) in 1987. Improved government support of nuclear energy was assigned to the American Nuclear Energy Council (ANEC), and public opinion issues were assigned to the U.S. Council for Energy Awareness (USCEA). These three organizations were merged into the Nuclear Energy Institute in 1994. Throughout this period, EPRI remained responsible for the design issues and for support of ARC on project activities.

3. THE U.S. NUCLEAR ENERGY INDUSTRY STRATEGIC PLAN FOR BUILDING NEW NUCLEAR POWER PLANTS; ACCOMPLISHMENTS UNDER THE STRATEGIC PLAN

In November 1990, the Nuclear Power Oversight Committee (NPOC) initiated the *Strategic Plan for Building New Nuclear Power Plants*. NPOC chartered the Ad Hoc Committee on the Strategic Plan to manage and update the plan each year. Since the formation of the Nuclear Energy Institute (NEI) in 1994, the Strategic Plan has been overseen by the NEI Executive Committee.

The concept behind the Strategic Plan was to integrate the industry's existing efforts related to ALWRs, and to address the emerging institutional and technical issues on which significant progress must be achieved to make nuclear energy a viable option for the future. The Plan:

(1) Identified the significant enabling conditions (technical, regulatory, environmental, financial, legislative, organizational, political and public acceptance) that must be met to achieve the goal;

(2) Assigned responsibilities to the appropriate industry organizations for achieving each condition;

(3) Fostered effective coordination between government and industry that pooled respective expertise and resources to achieve common goals.

The building blocks are outlined in Figure 1, which shows the title and industry group with primary responsibility for each block.

First and foremost, the 1990s have brought the completion of the advanced standard designs that have been the Strategic Plan's principal technical focus – including the 1,350 MWe General Electric

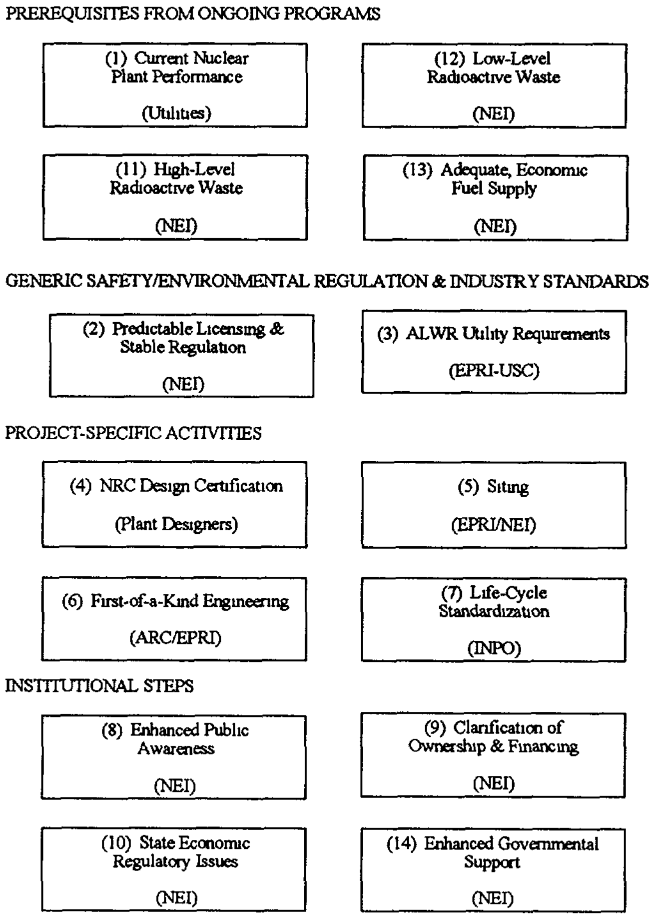

FIG. 1 Building block summary

Advanced Boiling Water Reactor (ABWR), and two advanced pressurized water reactors, the 1,350-MWe ABB Combustion Engineering System 80+ Standard Plant, and the 600-MWe Westinghouse AP600. These designs combine more than 40 years of industry experience in the design and operation of nuclear plants with the most exhaustive safety reviews ever performed by the NRC. Strong overseas interest—including purchases, commitments and other expressions of interest by a number of Asian nations—indicates that the superior safety, reliability and economics of these advanced designs is recognized worldwide and accepted as the basis for their continuing nuclear power plant programs.

Important developments are occurring in Asia. General Electric's ABWR design has been built by the Tokyo Electric Power Company at its Kashiwazaki-Kariwa site, and both units of this twin-unit station are in operation. The Korean Electric Company's Yonggwang two-unit plant, which became operational in 1995, incorporates many features of the ABB-CE System 80+ design; four other plants with additional System 80+ enhancements are under construction in Korea. The Taiwan Power Company has ordered a twin-unit ABWR from General Electric, and excavation of the Lungmen site as well as equipment fabrication and procurement are underway.

Through its Strategic Plan, the nuclear energy industry has achieved major accomplishments. Since the Plan was issued, the industry has steadily improved nuclear plant performance. For example, from 1990 to 1997, operating unit capability factors increased from a median of 71.7 percent to 81.6 percent. Average production costs have decreased—from 2.63 cents to 1.91 cents per kilowatt-hour. Safety performance indicators have also improved dramatically. These significant and steady trends have been impressive—increasing our confidence that improved, standardized nuclear power plants will compete favorably with other electricity generating options.

Congress passed major legislation, the National Energy Policy Act of 1992, which reformed the nuclear plant licensing process, committed $100 million to first-of-a-kind engineering for ALWRs, restructured the uranium enrichment enterprise, and directed improvements in the repository standards for disposal of the nation's spent nuclear fuel.

Key strategic plan milestones have been achieved in several areas of ALWR development in addition to the NRC final design approvals and design certifications discussed above:

(1) ALWR design requirements were developed by utilities, reviewed and approved by the U.S. Nuclear Regulatory Commission, and applied by reactor designers as the bid specification for standardized ALWRs.

(2) First-of-a-kind engineering—funded jointly by the Department of Energy at $100 million and by industry at $175 million—has been completed for the ABWR and will be completed in 1998 for the AP-600. This work achieves a high level of plant engineering design completion and provides critical data on the schedule and cost of construction, providing necessary certainty for improved project planning. The design and construction of the ABWRs for Taiwan Power Co.'s Lungmen project are making valuable use of this FOAKE work completed for the ABWR.

In addition, important progress has been made on a number of long-standing institutional issues, including enhancing policy-maker and public recognition of the need for nuclear energy; efficient management of low-level waste; passage of spent nuclear fuel legislation; assuring an adequate, economic fuel supply; and ensuring that policies of federal and state governments and the financial community recognize the total, long-term benefits of nuclear energy to the nation.

4. REMAINING TECHNICAL AND INSTITUTIONAL ACTIONS

The May 1998 update to the Strategic Plan to Build New Nuclear Power Plants is the final report of that Plan. Although the U.S. industry is shifting to a new strategic plan for the 21st Century, the 1998 Strategic Plan will remain as the detailed plan for ALWR actions completed and remaining. The industry remains committed to its original goal: to establish the necessary conditions for bringing about new nuclear plant orders in the United States. Conditions that are not yet fully in place, including institutional policies and practices conducive to nuclear energy, will be pursued as part of a more comprehensive industry strategy for positioning nuclear energy for the 21st century. Among the activities that will continue to be vigorously pursued are:

(1) complete the AP600 design certification;

(2) continue to improve plant operating and safety performance and continue to demonstrate that economic and safety improvements go hand in hand;

(3) continue to build support for nuclear energy and recognition of its benefits among political leaders and the general public as the major source of safe, reliable, emission-free electricity for meeting U.S. needs in the 21st century;

(4) continue to support license renewal for existing nuclear plants;

(5) ensure that state and federal policies shaping a restructured, competitive electricity marketplace reflect the importance of nuclear energy to the long-term national interest;

(6) stay the course on pressing for the necessary federal, state and local action to address spent fuel disposition and low-level waste management;

(7) continue to work with the NRC and Congress to ensure regulatory policies and practices do not unduly put nuclear plants at a competitive disadvantage relative to alternative baseload generating technologies;

(8) work with the NRC staff to establish appropriate emergency planning requirements for ALWRs and to develop common understandings of the Part 52 licensing process, including key issues related to licensing, construction verification and transition to start-up;

(9) increase emphasis on applying "spin-off" benefits of the Utility Requirements Document and the new ALWR designs to improve the safety, reliability and economics of existing plants;

(10) identify opportunities to further demonstrate the early site approval process;

(11) prepare to assist prospective owner/operators of future plants in the further development and regulatory acceptance of standardized operating processes and in the preparation and NRC review of license applications;

(12) continue efforts to assure an adequate, economic fuel supply to meet the needs of current and future nuclear plants in the United States;

(13) monitor and learn from ALWR construction and operating experience overseas to enhance planning for new plant orders in the United States.

Just as industry-government cooperation in research and development has been a key sub-plot to the progress made toward the goals of this strategic plan, expanding the scope and benefits of nuclear energy in the U.S. for the next century will require continued federal support. Most of the ALWR project-specific accomplishments under this plan could not have been achieved without a strong partnership between industry and the Department of Energy, with the strong support of Congress. While the industry funded about two-thirds of the total cost of this work, these resources would not have been made available without the cost-sharing and committed support of the federal government. The federally co-funded ALWR program was completed in 1997. Although no FY98 funding was provided for nuclear energy, key congressional leaders urged DOE to propose a new nuclear energy research and development (R&D) program for FY99.

Recognizing the strategic importance of nuclear energy to the nation, and the essential role of R&D to support continued advances in nuclear technology, DOE and industry worked together in late 1997 to produce the *"Joint DOE-EPRI Strategic Research and Development Plan to Optimize U.S. Nuclear Power Plants."* This joint R&D plan is based on common goals and objectives for nuclear energy R&D that have already been endorsed by industry and government. It expands those goals and objectives into R&D tasks to meet these needs. The focus of the EPRI-DOE plan is on currently operating nuclear plants in the U.S.—exploiting new technologies to further improve their economic and safety performance, and to extend their safe and economically useful life beyond current licensed operation. The plan will also support license renewal by ensuring the latest data are available to answer technical questions that might arise during NRC review.

The EPRI-DOE plan also lists future R&D goals and objectives related to further improving efficiency and reducing costs associated with ALWR designs. These goals and objectives focus primarily on improved construction technologies, application of the latest digital technologies, and similar enhancements. These activities are not currently receiving either industry or DOE funding, but are expected to be the focus of future resources.

In 1998, Congress appropriated funds for one of the two new nuclear energy R&D initiatives proposed by DOE – in funding the Nuclear Energy Research Initiative (NERI), which is focused on long-term R&D and innovative ideas in nuclear technology. A parallel proposal for a new program focused more on technologies for current plants (the Nuclear Energy Plant Optimization [NEPO] program) was not funded, but is expected to receive funding in FY2000.

5. THE FUTURE FOR NEW NUCLEAR PLANTS IN THE UNITED STATES

The transition-in-progress of the highly regulated electric utility industry into a competitive electricity marketplace is fundamentally changing the rules for determining the need for new generating capacity and how that need is to be met. For the next few years, existing generating plants are expected to provide an adequate supply of electricity in most regions of the United States. The electric power industry is significantly improving the output from existing plants, particularly its nuclear units, and is developing a variety of resources—conventional generating capacity (primarily gas turbine), demand-side management, conservation, and non-utility generation—that will support a growth rate in electricity demand of about 2 percent a year. This is within the range of most growth forecasts.

Soon after the turn of the century, a growing need for new baseload capacity is forecast to replace and augment the aging workhorses of the U.S. electric supply system. In 1970, 83 percent of U.S. baseload power plants were less than 20 years old; only 9 percent were over 30 years old. By 2000, only one-quarter of the baseload power plants will be less than 20 years old, while more than one-third—about 140,000 megawatts—will be over 30 years old. Some of this baseload capacity must be replaced as older plants reach the end of their economic life.

Moreover, the composition of the U.S. electric supply system is changing. In the 1980s, virtually all new generating capacity was baseload, and by 1990, the proportion of baseload capacity was above historical norms. As a result, much of the new capacity being built today is peaking capacity. By the year 2000, the proportion of peaking capacity will be at an all-time high, and the proportion of baseload will be near or below the historic norm. This suggests that the United States will need new baseload power plants in the next decade.

The industry believes that new nuclear plants will be selected to help meet demand for additional baseload capacity for several reasons:

(1) Emission limitations and "air pollution caps," such as those required by the amendments to the Clean Air Act, will increase the cost and potentially limit the ability to generate electricity from fossil-fueled plants in certain areas.

(2) Increased emphasis by policy-makers on actions to limit greenhouse gas emissions will result in a greater priority on generating plants that do not produce greenhouse gases. Some policy-makers are calling for an "emission-free portfolio" for new power generation additions that would maintain the existing U.S. percentage of electricity that comes from emission-free sources, including nuclear and renewable energy.

(3) There will be increased uncertainty regarding the price and reliability of supply and delivery of large quantities of natural gas for use in baseload power plants, as well as increased recognition that renewable energy alone, despite its popularity, will not be able to fill the gap in electricity demand.

(4) Today's more than 100 U.S. nuclear plants have an outstanding and upwardly trending record of performance, and the industry has numerous initiatives under way to further improve their operations. Extensive operating experience with today's plants and the promise shown in the ALWR Program provide a strong foundation for continued, and expanded, reliance on light water reactor technology.

(5) Experience from the construction of ALWRs in other countries will provide the foundation to proceed with new nuclear plant orders in the United States.

(6) A 1992 study by the nation's most prestigious scientific organization, the National Academy of Sciences, "Nuclear Power: Technical and Institutional Options for the Future," commended the R&D objectives of the ALWR Program. A study completed in 1997 by the President's Committee of Advisors on Science and Technology also underscored the benefits of nuclear energy to the nation and recommended substantially increased federal funding for nuclear energy R&D.

In the next few years, companies must start planning for new power plants to meet increased demand and to replace plants that reach the end of their operating lives. The intent of this plan has been to ensure that when new baseload generating plants are needed, the nuclear energy option will be available. The need for new baseload power plants early in the next century dovetails well with the significant progress made on all fronts under this strategic plan.

6. CONCLUSIONS

This decade also has brought successes and opportunities for nuclear energy: hard-won and remarkable improvements in U.S. nuclear power plant safety, reliability and economic performance; significant demand for U.S. nuclear technology overseas, including a growing interest in Asia in U.S. ALWR plants; a growing global awareness of environmental issues that makes nuclear energy an ever more compelling energy option; and the positive impacts of industry restructuring that are improving nuclear energy economics while maintaining high standards of safety.

As the environmental and energy policy goals of the nuclear industry and the nation begin to converge, the strategy for the 21st century must be to remove any remaining barriers and economic and political biases against the increased use of nuclear energy. This will facilitate license renewal for current plants and permit construction of new emission-free nuclear plants in the United States—specifically NRC-certified ALWR designs.

The industry's expanded strategic direction will take up this challenge. It will provide a compelling foundation for advocating the significant contribution of nuclear energy toward meeting the energy and environmental challenges of the next century. Emission-free nuclear energy will be the generating option of choice for new baseload capacity. A key bellwether for new nuclear plant orders will be industry experience with the NRC in renewing the licenses for existing plants.

Finally, the nuclear industry will encourage a new commitment to a farsighted national energy strategy—one that acknowledges that nuclear energy is essential to our future, and that invests in energy research and development consistent with that strategy. There is great potential for continued improvements in nuclear technology that will further enhance its safety, reliability and economics, while fulfilling its role as an emission-free source of electricity.

ADVANCES IN TECHNOLOGIES
RELATED TO NUCLEAR SAFETY

(Session III)

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ADVANCES IN TECHNOLOGIES FOR DECAY HEAT REMOVAL

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Abstract

The various decay heat removal concepts that have been used for the evolutionary water reactor plant designs developed worldwide are examined and common features identified. Although interesting new features of the "classical" plants are mentioned, the emphasis is on passive core and containment decay heat removal systems. The various systems are classified according to the function they have to accomplish; they often share common characteristics and similar equipment.

1 INTRODUCTION

Various water-cooled reactor concepts are at different R&D and design phases today. These include plant designs that incorporate well proven active safety systems, as well as plants where certain safety functions, and in particular long-term decay heat removal from the core and/or the containment, are achieved by passive systems. Examples of these evolutionary plants (evolutionary Light Water Reactors, LWR or Heavy Water Reactors, HWR) are listed below, in three categories: Pressurized Water Reactors (PWR) of Western and Eastern design, Boiling Water Reactors (BWR) and Heavy Water Reactors (HWR):

LWR/PWR: EPR, System 80+, KNGR, APWR, EP 1000, AC-600, AP-600, MS-600, etc. Evolutionary VVER-1000, VVER-640, etc.

LWR/BWR: ABWR, BWR 90, ESBWR (SBWR), SWR-1000, etc.

HWR: CANDU-9, CANDU-6, AHWR, etc.

Design information on these systems can be found in recent IAEA-TECDOCs [1]-[4] and in the proceedings of this conference. The *phenomena* that are relevant for decay heat removal and their importance are discussed in [5].

1.1 Core and containment cooling

In water reactors, the fuel bundles must be kept covered with water to ensure their coolability. During the first phase of Loss of Coolant Accidents (LOCA), the primary coolant systems undergo complex transients. The main concerns during this phase are the evacuation of the heat *stored* in the fuel rods during normal operation and retention of sufficient water in the Reactor Pressure Vessel (RPV). This is achieved by keeping adequate thermal-hydraulic conditions in the core and replenishing the coolant lost during the blowdown process. In this respect, the presence of larger water inventories in the RPV during normal operation are beneficial. At the end of this first phase, the state of the primary system is stabilized; the main concern then becomes the evacuation of the decay heat. The decay heat must be removed first from the core in the RPV and from the primary system, and then from the containment where typically it is finally dumped.

This evacuation of the decay heat has been assured in most "classical" water reactor systems by redundant and diverse *active* Emergency Core Cooling Systems (ECCS) and containment cooling systems. The most recent ALWR and AHWR designs take advantage of accumulated experience and combine the best characteristics of existing reactor systems in an optimal way to achieve even higher reliability and safety in active core and containment heat removal. Examples are the EPR, ABWR, BWR 90, System 80+, KNGR, etc. [1], as well as evolutionary CANDU and VVER designs. High degrees of reliability and safety can be achieved by increasing system redundancy, separation, diversity, etc. Since active systems need fairly large power supplies to operate, the availability of the sources of electricity must also be improved. Such improvements bring, however, added complexity to the systems.

1.2 The passive plants

In certain other new-generation evolutionary plants designed worldwide, attempts have been made to reduce the complexity of the long-term decay heat removal systems; one approach adopted in several designs has been to achieve this safety function via increased use of *passive* systems.

The passive plants require no operator actions to mitigate Design-Basis Accidents (DBA). Passive systems use only "natural" forces such as gravity, natural circulation and compressed gas to operate. Containment structures, water pools or the atmosphere provides the heat sinks needed to dispose of the decay heat. There are no *active* components such as pumps, fans, diesels, water chillers, etc. Passive systems may require, however, the alignment and actuation of a few valves; passive valve actuators have also been proposed, but they may not be indispensable, given the relative simplicity and the high degree of reliability that simple valve alignments can achieve. Thus, passive systems do not require redundant, *safety-grade*, active ECCS and containment cooling systems and the corresponding redundant safety-grade emergency power supplies. The ambient air is most often the ultimate heat sink; this results in the elimination of the safety-grade service water system. The elimination of safety-grade systems should result in considerable simplification of the plants and capital cost reductions. In addition, typical unattended operation periods of the order of days (typically 72 hours) can be achieved.

The classical ECCS and containment cooling systems are typically replaced by:

- Natural-circulation cooling of the core (when the primary system is intact)
- Gravity Driven Cooling Systems (GDCS) (with the primary system breached)
- Passive Containment Cooling Systems (PCCS)

Since the pressure and temperature differences driving passive cooling are usually small, the corresponding single-phase heat transfer rates are relatively weak and evaporation or condensation of the coolant is usually necessary to get reasonable heat transfer areas and heat exchanger sizes.

This paper reviews design approaches taken to improve the safety of long-term heat removal from the core and the containment in the new generation of evolutionary plants, in particular the plants described in [1]. Certain advances achieved by further improving *active* systems and their configurations are mentioned, but the main emphasis is on reviewing the various *passive* approaches proposed and in identifying common features and trends. The paper does not consider the so-called *innovative* plants where the design deviates significantly from that of existing plants and where more radical approaches have been proposed [1].

The various systems are classified in terms of the *function* that they have to accomplish. In this light, one finds out that although there are many combinations of possible passive systems and their variations, they most often share common characteristics.

2 DECAY HEAT REMOVAL FROM THE CORE AND THE CONTAINMENT

After certain incidents or accidents, avoidance of further degradation of the system requires:

a) Management of the condition of the *primary system:* the core must be kept covered, *and* the decay heat removed from the primary system. Keeping the core covered requires refilling of the RPV, classically via the ECCS. If the decay heat is not fully removed from the primary system via the break (case of small primary-system breaches), the primary system must either be forcefully depressurized (the classical solution in BWRs) or the Steam Generators (SG) must also be engaged in the decay heat removal process (PWRs).

b) Evacuation of the decay heat from the *containment.*

In the **"classical" new evolutionary plants,** improved decay heat removal capability and safety are achieved by *active* systems having:

- even better system design: greater redundancy, independence, separation, etc. (typically four separate trains according to the n+2 redundancy concept)
- other improved characteristics such as:
  - a larger water inventory in the RPV (or in the SG secondary side): this results in later core uncovery (or SG dryout) [GY3]
  - larger pressurizer volume [GY4]
  - elimination of primary system piping (to decrease the probability of a LOCA [GY5])
  - Direct Vessel Injection (DVI) of emergency coolant [GY6]
  - relocation of emergency cooling water sources inside the containment:[GY7] e.g., relocation of the Refuelling Water Storage Tank (RWST, that provides emergency coolant) inside the containment: the In-Containment RWST (IRWST) solution
  - flooding of the reactor cavity (to a level above the top of the fuel [GY8])
  - automatic depressurization of the primary system followed by low-pressure safety injection
  - use of fire water system for cooling, e.g., as containment spray, etc.

The ***passive*** **plants** incorporate novel technologies; these are *the focal point in this review.* Passive plant features will be discussed according to system function or according to the different accidental plant conditions and corresponding passive approaches for decay heat removal. The states or cases considered below are:

- Primary System intact but loss of the heat sink (SG or turbine)
- Primary System breached at high or medium pressure
- Primary System breached and depressurized
- removal of decay heat from the Containment.

3 REMOVAL OF THE DECAY HEAT FROM AN INTACT PRIMARY SYSTEM

If the Primary System is intact but the normal heat sink (secondary side of SG or turbine) has been lost, the decay heat must still be removed from either the RPV (BWR) or the SG (PWR). The following solutions, all based on the *passive connection of the primary system to a Heat Exchanger (HX)* (or condenser) have been proposed:

- *Heat exchangers connected to the primary system and immersed in a water pool inside the containment.* Examples are the AP-600 or the EP 1000, Figure 1, where a Passive Residual Heat Removal (PRHR) HX is immersed in the IRWST. A similar, but limited-capacity solution for PWRs is also considered in [6]. The SWR-1000 has Emergency Condensers permanently connected to the core and located in the Core Flooding Pool, Figure 2. Residual heat removal in these cases is a *two-step process*, since the pools eventually saturate and vaporize and the steam must be condensed by another system.
- An alternative is the *cooling of the secondary-side of the SGs* using a condenser. Such immersed emergency condenser solutions have been adopted for the KNGR [7], for CANDU systems (Figure 3), and by Siemens. The VVER-1000 and the AC-600 use natural-circulation air-cooled condensers located outside the containment. The air-cooled solutions provide an unlimited heat sink at the likely expense of a very large heat transfer area.
- A solution involving *Isolation Condensers connected to RPV* and immersed in external pools has been adopted for the ESBWR, Figure 4, and is also used for the Indian, heavy-water-moderated, light boiling-water-cooled AHWR.
- A similar solution has been adopted for the passive cooling of the moderator in CANDUs, Figure 3.

4 DECAY HEAT REJECTION IN CASE OF A LOCA

Several new designs have been improved by placing emergency cooling water sources *inside* the containment. For example, in the APWR the refuelling water storage tank has been moved inside the containment, as noted above. The AP-600 and the EPP have several water sources located inside the containment: Core Make-Up Tanks (CMT), high-pressure accumulators, lower-pressure Core Reflood Tanks (CRT) and also an In-Containment Refuelling Water Storage Tank (IRWST).

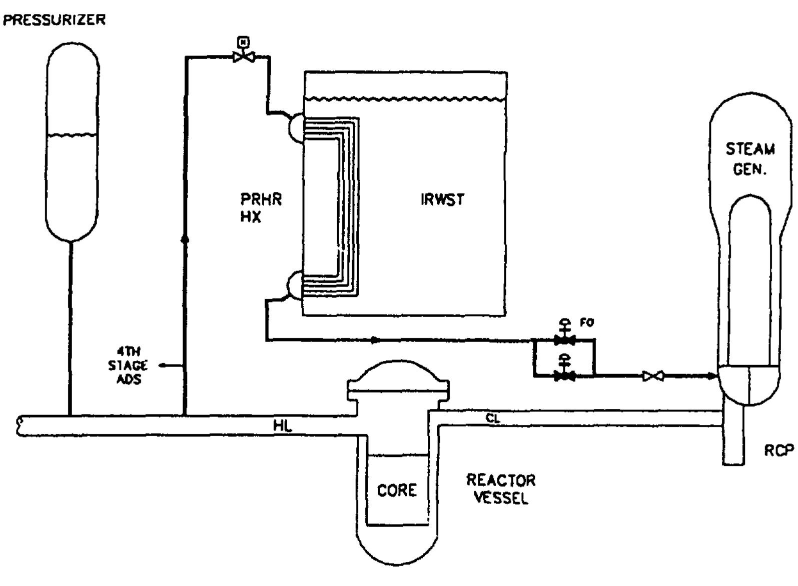

FIG. 1. The AP-600 Passive Residual Heat Removal (PRHR) system using a HX connected to the primary system and immersed in the IRWST.

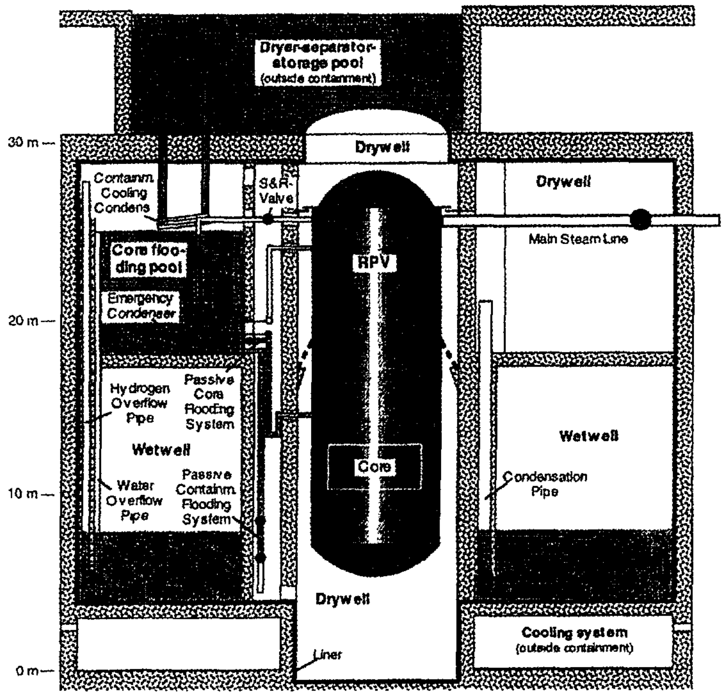

FIG. 2. The SWR-1000. Scramming of the reactor leads to collapse of the voids in and above the core region; this leads to automatic activation of the Emergency Condenser connected to the RPV without valves and immersed in the Core Flooding Pool. After depressurization, the Core Flooding Pool provides gravity cooling to the primary system. The Containment Cooling Condensers condense steam in the containment; light non-condensibles that may accumulate near the roof of the containment are vented to the Suppression Pool.

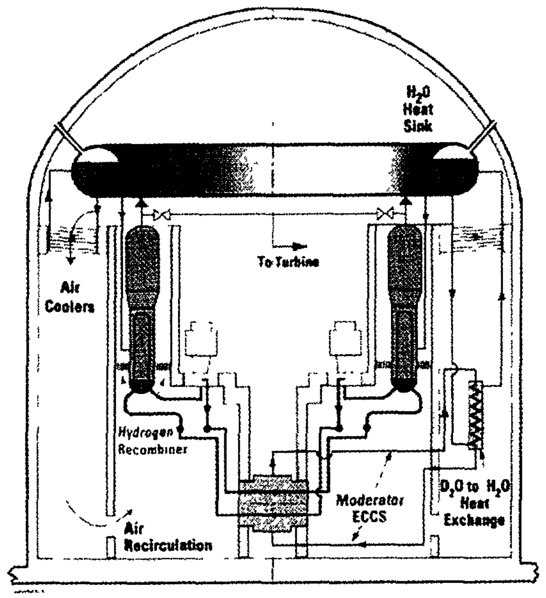

FIG. 3. Evolutionary CANDU 6 passive heat-rejection systems. The Steam Generator Heat Rejection system condenses steam from the SG in a condenser located inside the large toroidal Passive Emergency Water System (PEWS) tank. The Moderator Heat Rejection System has an intermediate HX also rejecting heat to the PEWS. The containment is cooled by air coolers fed by water from the PEWS that promote natural circulation inside the containment. The PEWS acts as the heat sink for all these systems.

To eliminate the need for long-term continuous pumped addition of coolant to the vessel, in several new designs there are provisions for flooding the reactor cavity to a level above the top of the fuel. Examples are the BWR 90 and most of the passive plants.

The primary system of LWRs is designed so that the core can be kept covered in spite of breaches in the primary system. Elimination of primary system piping contributes, however, also to the elimination of certain LOCA scenarios. Examples are the elimination of the recirculation piping in the ABWR by use of Reactor Internal Pumps. A somewhat similar trend can be observed in the AP-600 and the EPP where the primary system recirculation pumps were directly attached to the SGs.

Considering the continuum of breaches ranging from "intact primary system" to large-break LOCA, one realizes that decay heat removal from the primary system under high pressure can partly be performed by the systems for removal of the decay heat from an intact primary system discussed above. Thus, only additional solutions proposed for make up of the primary inventory and for medium and low pressure emergency injection are mentioned below.

The AP-600 uses a *Core Make-up Tank (CMT)* The pressure on top of this tank is equalized with primary system pressure, Figure 5. Thus, the CMT can provide make-up water to the core by gravity at any pressure. A CMT is also used in the AC-600.

For intermediate pressure levels in PWRs, injection of water from *accumulators* (at about 5,0 MPa (50 bar)) or core reflood tanks (at about 1,5 MPa (15 bar)) is used.

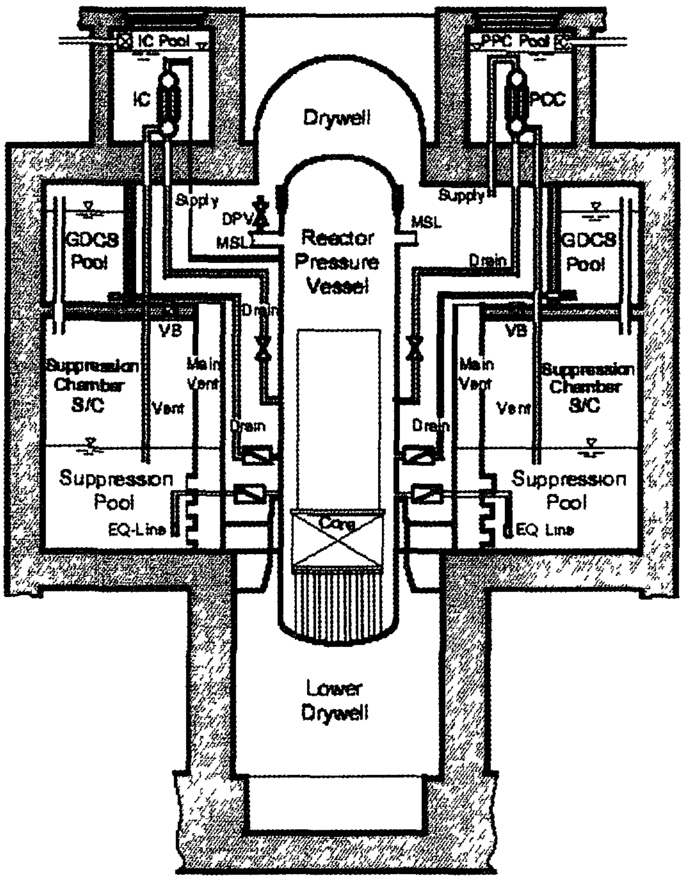

FIG. 4 The ESBWR passive core and containment cooling systems The Isolation Condensers (IC) condense steam from the RPV. The Gravity Driven Cooling System (GDCS) pool floods the core after depressurization of the primary system The Passive Containment Cooling System (PCCS) condenses containment steam and vents the non-condensibles to the Suppression Pool.

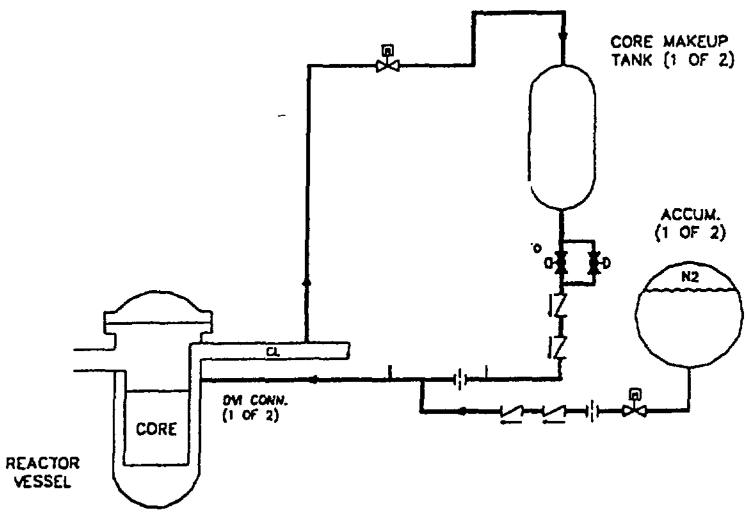

FIG. 5. The principle of the Core Makeup Tank, as used in the AP-600: the top of the tank is connected to the primary system by a pressure equalization line; thus, gravity injection of coolant into the RPV at any pressure is made possible.

To better cope with the pressurized LOCA scenarios, two approaches have generally being used:

a) intentional automatic depressurization of the primary system (similar to the Automatic Depressurization System, ADS, of the classical BWRs) and subsequent use of low-pressure safety injection (LPCI) systems, or

b) increase of the capacity of the high-pressure coolant injection (HPCI) system.

ADS systems have been incorporated in the AP-600 and the EPP. Other designers have opted, however, for higher-capacity LPCIs; for example, the ABWR has a reinforced HPCI relieving reliance on its ADS.

In the passive plants, one relies on *automatic depressurization* of the primary system and actuation of low-pressure gravity-driven core make-up systems. This solution is retained for the ESBWR (Figure 4) and the SWR-1000 (Figure 2). Both passive BWRs provide for gravity-driven, low-pressure core flooding. The AP-600, after depressurization, uses the IRWST inventory to reflood the RPV by gravity.

5 REMOVAL OF THE DECAY HEAT FROM THE CONTAINMENT

All containment systems profit from the *passive heat sink* provided by the *structures* inside the containment and the containment walls. The structures are usually needed to absorb the higher level of decay heat generation immediately after shutdown and limit the initial containment pressure; by the time these heat sinks get "saturated" (reach equilibrium temperatures with the containment atmosphere), the decay heat levels are lower and other containment cooling systems take over the decay heat removal function. Thus, the needed capacity of containment cooling systems is reduced.

Novel solutions that have been proposed for containment cooling include:

- Cooling of the containment building in the AP 600 from the outside by natural draft enhanced by a water film on the wall, Figure 6. Such solutions are possible with metallic containment walls only.

- An Italian alternative solution [8] for the EP 1000 proposes a finned condenser installed near the roof, inside the containment building, an intermediate sealed thermosiphon loop penetrating through the double concrete containment walls, and an external hybrid (initially immersed, water-cooled and later air-cooled) HX, Figure 7. Cooling of the containment atmosphere by *containment condensers* installed near the roof is also proposed for both the SWR-1000 and for CANDU systems: the SWR-1000 has a containment-cooling condenser with its secondary system connected to an external pool, Figure 2. The CANDU 6 containment coolers have their secondary sides connected to the PEWS tank, Figure 3.

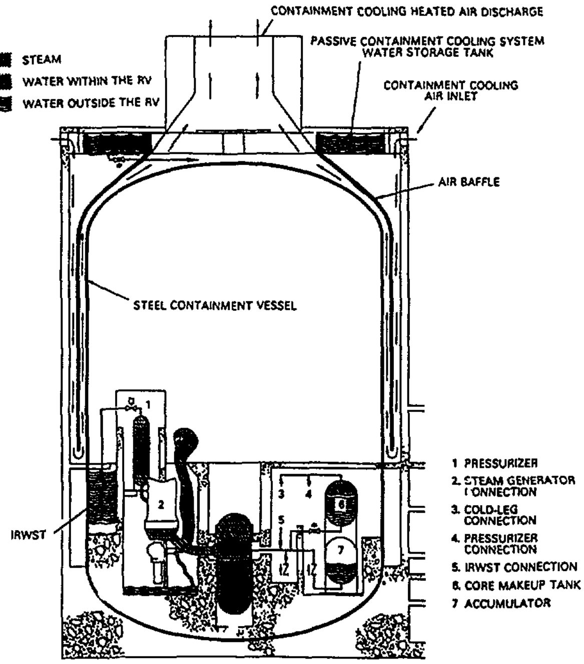

FIG. 6. Passive containment cooling for the AP-600. Air circulates by natural draft in the space created between the metallic containment wall and the outside concrete wall. The water storage tank at the top of the building wets the containment surface with a water film, needed to enhance the process after shutdown, when the decay heat is still high.

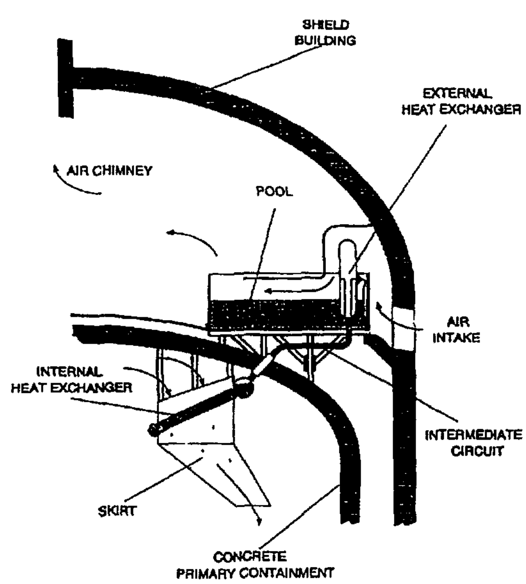

FIG. 7. Passive containment cooling solution proposed as an alternative to the AP-600 external-wall cooling concept.

- Finally, the ESBWR proposes a novel Passive Containment Cooling System (PCCS): Condensation of containment steam takes place in a *condenser immersed in an external pool*. The condenser tubes are always connected to the drywell. The noncondensibles are discharged to the Suppression Pool by an ingenious venting system, Figure 4.

6 CONCLUSIONS

A large number of evolutionary LWRs and HWRs with improved primary-system and containment decay heat removal concepts have been proposed. These have been categorized and presented summarily in this paper, according to their intended function. Although there is a diversity of designs stemming from all countries developing evolutionary water reactors, there is a relatively small number of optimal technical solutions that have been retained. Several novel passive cooling solutions have been proposed and constitute the main emphasis in this review.

Most systems rely on boiling and condensation to obtain sufficiently high heat transfer rates under natural circulation conditions. With the primary system intact, decay heat is removed by circulating the primary coolant in heat exchangers or condensers, typically immersed in pools. Novel solutions for decay heat removal from the core rely on depressurization of the primary system followed by flooding of the core by gravity or with high-pressure gravity-driven core make-up tanks connected at their top to the primary system. Novel solutions for decay heat removal from the containment are based either on cooling of the (metallic) containment wall from the outside, or on use of condensers; these can be located either inside the containment near the roof (the containment steam condensing on the outside of the tubes) or outside the containment, immersed in pools (the containment steam condenses inside the tubes).

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CONTAINMENT: MORE THAN ONE WAY TO MEET SAFETY AND ECONOMIC GOALS

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Abstract

The defense in depth approach will continue to be the basis for sound design for future plants. Moreover, it will be strengthened by additional margins in the design considering of severe accidents with a realistically conceivable approach. The containment system is a key element of the strategy and it is designed to provide a last barrier to the release of radioactive material into environment following release internally in the severe accidents despite all efforts to prevent such accidents. Therefore, in comparison with the present practice, clear improvements may be sought. However, due to the different current licensing practices, rules and regulations, the degree of the improvements and the preferred containment type may vary from country to country. This paper discusses major factors affecting containment design considering both the current and extended design bases due to the postulated severe accidents. An overview of advances for the newly developed reactors is also surveyed and addressed, especially for the containment system design features.

1. INTRODUCTION

Containment is a key component of the mitigation parts of the defense in depth philosophy of a nuclear power plant since it is the last barrier designed to prevent large radioactive material release into the environment in the event of an accident. In addition, because the containment is contributing a significant part of the entire station cost, considerable efforts have been devoted to reduction of the costs associated with construction and maintenance.

While the current containments meet the objective they have designed for and provide a substantial contribution to the defense in depth approach, many design improvements are proposed for future plants to further enhance safety and economic aspect.

This paper discusses the design bases based on the imposed regulatory and design requirements for the containment, and describes the major design factors which influence the containment design. This paper also surveys the different proposed containment designs of the advanced reactors.

2. CONTAINMENT DESIGN BASES

The most important function of the containment system is to prevent the release of radioactive fission products to the environment within the acceptable level stipulated by the authorities. The containment system must be designed to withstand the imposed loads due to internal and external challenges.

The safety for currently operating reactors is categorized by three levels of design approaches, based on the current licensing and design basis requirements:

- Accident resistance is ensured by design margins, redundancy and diversity of safety systems, and in-service inspection and testing to assure reactor coolant pressure boundary integrity
- Core damage prevention is ensured by dedicated safety systems to meet regulatory requirements
- Mitigation is provided by the containment and associated systems

During the last decade, in the aftermath of the TMI and Chernobyl accidents, the problems related to severe accidents – accidents beyond design basis events – have been discussed. Owing to the inherent phenomenological uncertainties limiting a complete understanding of the extremely complex severe accident scenarios, the two ultimate conclusions of the research are:

- Nuclear plant designs must incorporate engineered safety features intended to prevent the occurrence of a severe accident and to mitigate its consequences if one were to occur
- The containment structure must be robust to safely withstand the consequences of a severe accident with ample design margins

For the advanced reactor, the design requirements are extended so that the containment integrity shall be warranted for all the accidents including the postulated severe accidents involving core damage. The severe accident challenges are;

- High Pressure Melt Ejection (HPME) and Direct Containment Heating (DCH)
- Hydrogen generation and combustion
- In-vessel and Ex-vessel fuel-coolant interaction (FCI)
- Molten core-concrete interaction (MCCI) and possible basemat melt-through
- Mass and energy releases to the containment causing high pressure or high temperature
- Containment by-pass

To cope with severe accidents, the ALWR adopts new system design measures such as reactor cavity flooding system, improved RCS depressurization system, hydrogen control system, and enlargement of reactor cavity area, etc. Furthermore, the containment shall be designed robust to withstand the consequences of a severe accident with ample design margins, and large containment free volumes are an absolute requirement for the ALWR because of the severe accident concern with the generation of hydrogen.

The containment should maintain its role as a reliable leak-tight barrier by ensuring that containment stresses do not exceed the ASME Service Level C limits for steel containments (or its equivalent for concrete containments) for a minimum period of 24 hours following the onset of core damage to provide a competent barrier against the uncontrolled release of fission products.

To minimize the overall risk of containment melt-through, the reactor cavity will be arranged with a sufficient floor area and a configuration to prevent from depositing large quantities of energy directly into the containment atmosphere and thereby rapidly pressurizing the containment, and equipped with a cavity flooding system, or other mitigation features.

Based on the operating experience over the past three decades, many utilities started to specify their special desires for new nuclear power plants, which results in the development of utility requirements for future advanced reactor concept. USA, Europe, Japan, Korea and Taiwan developed utility requirements document to present a clear, complete statement of utility desires for design, construction and performance of an advanced reactor.

Major commonalties related with containment design are:

- A design life time of 60 years
- Core damage frequency less than 10E-5 per year and cumulative frequency of large releases following a core damage less than 10E-6 per year
- Design measures to cope with severe accidents

With respect to dealing with licensing procedure and/or the impact of the new reactor to the environment, the requirements in Europe appear to be different from those in the USA as follows:

- To limit emergency protection actions to a minimum beyond exclusion area boundary (EAB) during early releases from the containment
- To avoid delayed actions (temporary transfer of people) at any time about 3 km from the reactor
- To avoid long term actions, involving permanent resettlement of the public beyond EAB
- To ensure that restrictions on the consumption of foodstuff and crops will be limited in terms of time and ground area.

3. FACTORS AFFECTING CONTAINMENT DESIGN

3.1. Energy relationship

When sizing a containment for a particular nuclear steam supply system, it is necessary to estimate the most severe pressure-temperature history in the containment volume and containment structure that may possibly occur following a postulated accident. There are several significant factors influencing the containment design pressure and volume. The total energy in the containment following the accident is the sum of the energy added plus the energy initially present. The containment design pressure is inversely proportional to the containment volume and directly proportional to the energy released into the containment as a result of the postulated accident. The energy release into the containment is determined by the thermal rating of the reactor and the efficiency of the emergency core cooling systems (ECCS). For a given energy release, the optimum containment design pressure is then determined by appropriate cost studies. The cost will increase as the containment design pressure increases and the cost will decrease as the energy density increases, as shown in Figure 1. From this relationship, it may be seen that ideally a minimum containment cost exists for a given reactor.

3.2. Nuclear steam supply system and mechanical requirements

Another factor in the sizing of a containment is the physical size requirement to accommodate the nuclear steam supply system and all related equipment, which is a major parameter to determine the minimum containment diameter needed to accommodate all the necessary equipment in a functional manner. For a typical PWR containment, starting from the center and going toward the containment wall, space must be provided for the reactor pressure vessel, the primary shielding, steam generators, pumps and pressurizer, the concrete missile shield or secondary shielding, and auxiliary equipment such as fan coolers and tanks. The minimum height is determined by the crane hook elevation to assure that all equipment can be erected, moved during refueling, and repaired if necessary.

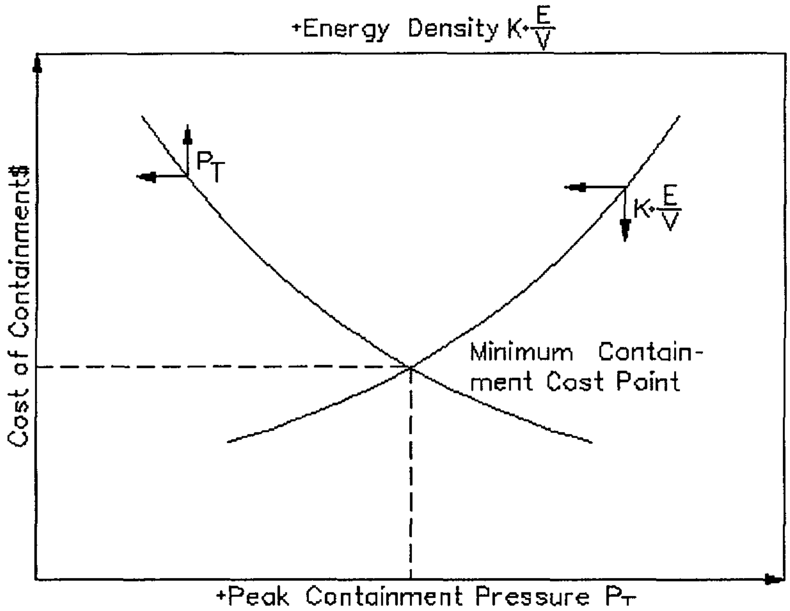

FIG. 1. Idealized Minimum Containment Cost Curve.

3.3. Site considerations

Site characteristics have a profound influence on the containment design. Earthquakes of intensities will influence the design of the reinforcements in the base slab and cylinder, and the configuration of the structure including the adoption of a common base slab. For instance, a lower containment with larger diameter experiences smaller seismic forces than a higher containment with smaller diameter, and the higher seismic intensity and the resulting seismic forces may influence the designer to change the containment type.

The soil characteristics influence the seismic analysis through the soil structure interaction. They influence not only the seismic forces but also the shape of the response spectra for which the structure and equipment has to be designed. Depending on the soil, the base rock excitation may be dampened or amplified across the overall response or just for certain frequencies. This is one reason for initiating the soil investigation very early in the plant design. In order to have the standard design enveloping the soil conditions, the System 80+ and the KNGR are examples performed the soil structure interaction analyses for the different soil conditions.

The baseslab design is influenced by the soil characteristics, especially the bearing capacities. For soil with low bearing capacities, piling or soil improvement may be required, and larger & thicker baseslabs may be designed to reduce the unit bearing pressure.

Another item related with reactor siting is an aircraft impact. In most cases thecontainments for nuclear power plants located within a certain range of airports are required to be able to withstand the impact of a postulated aircraft. The EPR is an example assumed to perform the aircraft crash defined by GRS/RSK from Germany.

The last major item is missile and explosives overpressure protection for primary containment structures. In regard to tornado-generated missiles, the most probable maximum tornado must be evaluated, and the analysis is required to assure that the postulated missile would not impair the ability of the plant to shutdown and be maintained in a safe shutdown condition. Other similar areas of concern are related to the location of missile sites and military installations in the vicinity of nuclear plants.

3.4. State of the art of design and construction

The types of containments can be categorized for the following:

- Pre-stressed concrete single containment (with steel liner plate)
- reinforced concrete single containment (with steel liner plate)
- cylindrical steel
- spherical steel
- pre-stressed concrete double containment (with and without steel liner plate)

The type and material selection of the containments have been taken with conservative approach. The ASME Code on steel pressure vessels and the leak-tightness of welded vessels influenced engineers to design welded steel plate containments in the early years of development.

As the capacity of the plant becomes larger, the containment was changed from steel to concrete containment. The Japanese selection of pre-stressed concrete containment for 4-loop plants is an example; for an 1100 MWe plant, the capacity of a cylindrical steel containment requires post-weld heat treatment (PWHT) of the steel vessel since it requires a plate thickness exceeding 44.5 mm which is the guideline of the PWHT specified in the Code.

The pre-stressed concrete containment has been developed in many areas as follows: tendon capacities have grown to 15345 kN, which may be applied to the EPR containment; tendon lengths have been increased from 150 feet to 600 feet which enabled the adoption of the hemispherical dome; the number of buttresses has decreased from 6 to 2 whose typical examples are Japanese Ohi 3&4.

Dome steel liner plate can be designed to act as a form during construction, and can be installed using a prefabricated module concept with containment spray piping installed which greatly helps the construction. The use of sliding form or jump form has been widely used for concrete containment construction.

For the medium size reactor, cylindrical steel containment is developed with passive design features because the steel vessel provides the heat transfer surface, which removes heat from inside the containment and ejects it to the atmosphere using only natural forces such as natural circulation.

Since the release limits from the plants are very stringent, most of the advanced reactors are designed with a double containment, which gives credit to decontamination factors for the radio-nuclides that escape the primary containment. In the EPR, credit is given for a grace period of several hours to the secondary containment for collection, deposition and hold up of fission products leakage through the primary containment shell.

The French double containment concept of the N4 or P4 Series consists of an unlined pre-stressed concrete primary containment enclosed by a reinforced concrete secondary containment, and the EPR may follow the same concept. The leak-tightness requirement of less than 1 % per day is the basis of design without provision of a steel liner, whereas the integrated leak rate for the typical PWR containment with steel liner has been 0.1 % per day.

3.5. Containment performance during postulated severe accident challenges

During the KNGR design development, a brief comparative evaluation of the primary containment with and without steel liner was performed based on the KNGR design parameters such as inside diameter of 45.7 m, design pressure of 414 kPa(g), severe accident pressure of 709 kPa(g). For the evaluation, unlined concrete containment is assumed to have a design pressure of 709 kPa(g) to meet the No-Concrete-Tension criterion under severe accident pressure. According to the evaluation, there should be a significant increase in the amount of pre-stressing required to provide full pre-stress for the unlined containment, whereas insignificant changes may be necessary for the lined containment.

According to the System 80+ containment analysis, spherical steel containment is also robust against severe accident pressure and temperature.

3.6. Economic considerations

Determining the type or size of existing containment for a particular plant usually begins after all of the below mentioned parameters have been established:

- The nuclear steam supply system
- The minimum size of containment required
- The volume-pressure relationship for the design basis accident
- The containment performance during a postulated severe accident
- The intensity of the earthquake
- The dynamic and static soil characteristics
- The soil bearing capacities
- The extent to which state-of the art construction should be implemented
- Target date of completion
- Cost evaluation

To arrive at the most economical containment for the particular design conditions, preliminary designs, material takeoffs, and cost estimates have to be made. A comparison of total costs and estimated construction time and construction manpower demand will point to the most economical solution.

4. ADVANCED DESIGN DEVELOPMENT ACTIVITIES ON CONTAINMENT

Advanced LWRs (ALWR) are developed along two lines: large units of 1000- 1500 MWe, and medium size of about 600 MWe. The large units are generally designed closely on existing plants, whereas the medium size units introduce new features such as passive safety systems and plant simplifications, aimed at defeating the economic disadvantage of their smaller size.

Some examples of large, evolutionary ALWRs are: the ABWR, the APWR, the EPR, the System 80+, and the KNGR. Among the medium size ALWRs, the AP-600 is a typical example of the design. The ESBWR and the EP 1000 are examples of large size reactors equipped with passive systems. New versions of HWRs have also been developed. In Canada, CANDU-9 is under development within the evolutionary program.

4.2. ABWR, GE USA in cooperation with Hitachi and Toshiba, Japan

The design of the ABWR represents a complete design for a nominal 1300 MWe power plant and realized with the construction of Kashiwazaki-Kariwa plant 6&7, which were taken into commercial operation in 1996 and 1997 respectively.

As shown on Figure 2, the ABWR pressure suppression primary containment, which comprises the drywell, wetwell, and supporting systems, is surrounded by the reactor building, which serves as a secondary containment. A negative pressure is maintained in the reactor building to direct any radioactive release from the containment to a gas treatment system.

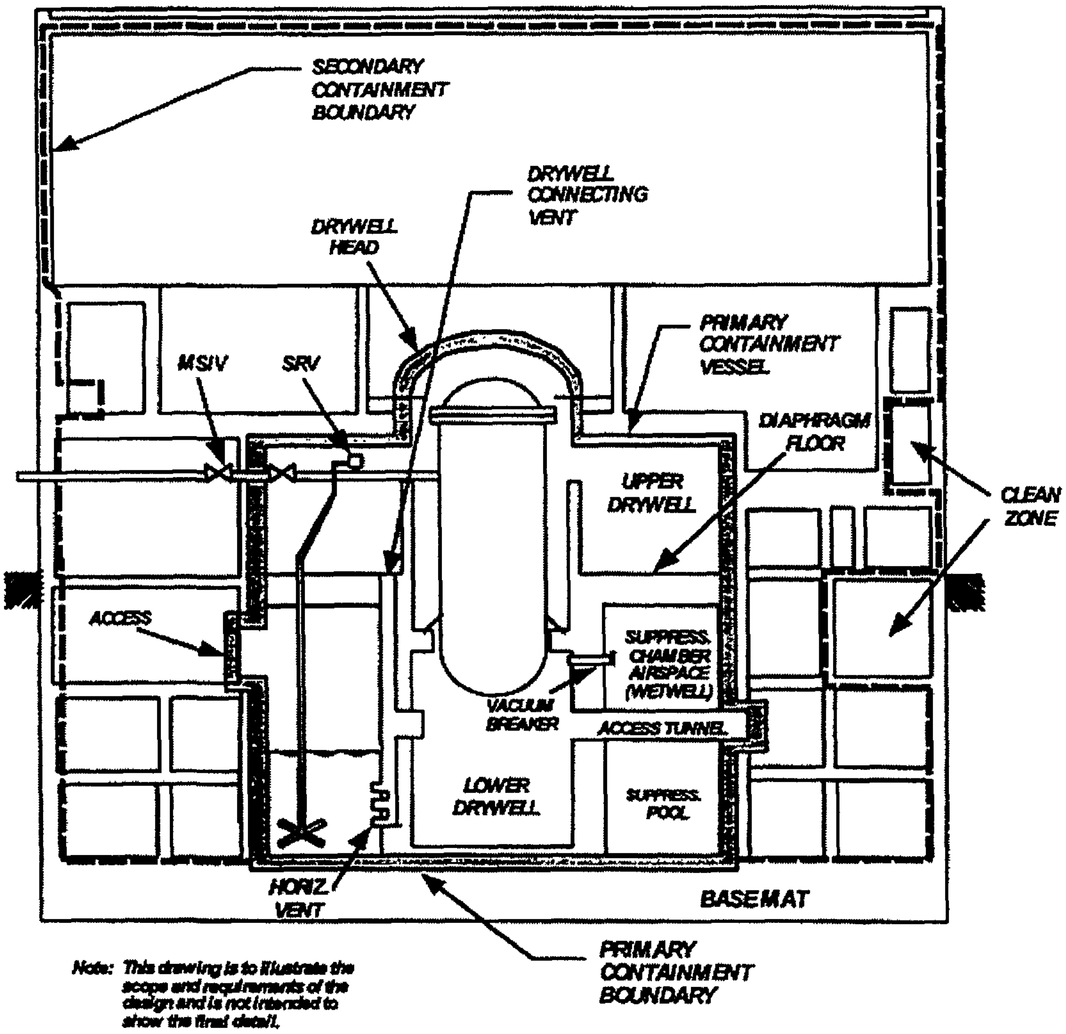

FIG. 2. ABWR – Containment structure.

The ABWR reactor building is a reinforced concrete structure. The integrated reactor building and containment structure was analyzed for a safe shutdown earthquake of 0.3g. Based on an assumed containment leak rate of 0.5%/day, the off-site doses after accident is less than 1 rem.

The design features of the ABWR containment system are shown in Table 1.

4.3. APWR, Mitsubishi, Japan/ Westinghouse, USA

The APWR has been developed for future use in Japan as a part of Phase III of the Improvement and Standardization Program of the Ministry of International Trade and Industry.

Even though the design of APWR containment has not been finalized, basic design was developed based on the Japanese PWR 4-loop plants of 1100 MWe as shown on Figure 3.

Major changes in containment design compared with the existing plants are:

- IRWST is adopted to eliminate the operation of changing the suction from the refueling water tank to the containment re-circulation sump which is needed during an accident on existing plants

TABLE 1. ABWR CONTAINMENT SYSTEM DESIGN FEATURES

	Design Feature	Purpose/Description
Containment	Primary containment Pressure suppression/ reinforced concrete Cylindrical shape Secondary containment provided?	Dimension (diameter/height)*: m Free volume*: m^3 Design pressure/temperature: 310/171.1 kPa/°C (severe accidents)*: kPa/°C Yes
Containment Heat Removal	Suppression pool Drywell spray system	Heat sink and pressure suppression for post-accident Containment heat removal by decay heat removal system
Hydrogen Control	Inerting system	Prevents hydrogen and/or oxygen combustion and detonation
Fission products Control	Suppression pool Standby gas treatment system	Fission product scrubbing and retention for post-accident Filter fission product leaked from primary containment – minimize offsite dose
Containment Isolation and leakage rates	Minimum penetration Containment leakage rates	Containment isolation with minimum leakage 0.5 vol %/day
Containment Overpressure Protection	Containment Overpressure Protection System	Prevent catastrophic containment failure and provide maximum fission product scrubbing with passive hard-piped wetwell vent controlled by rupture disks set at twice design pressure (service level C)
RCS Depressurization	Automatic depressurization system	Depressurize reactor pressure vessel and prevents high-pressure core melt. Minimize probability of DCH
Reactor Cavity Configuration	Basaltic concrete Lower drywell floor Passive drywell flooding system(fusible valve)	Prevents core-concrete interaction Provides sufficient spreading area for cooling of molten core Provides coolability of postulated core debris on the drywell floor.

Parameters that are not known or will be determined

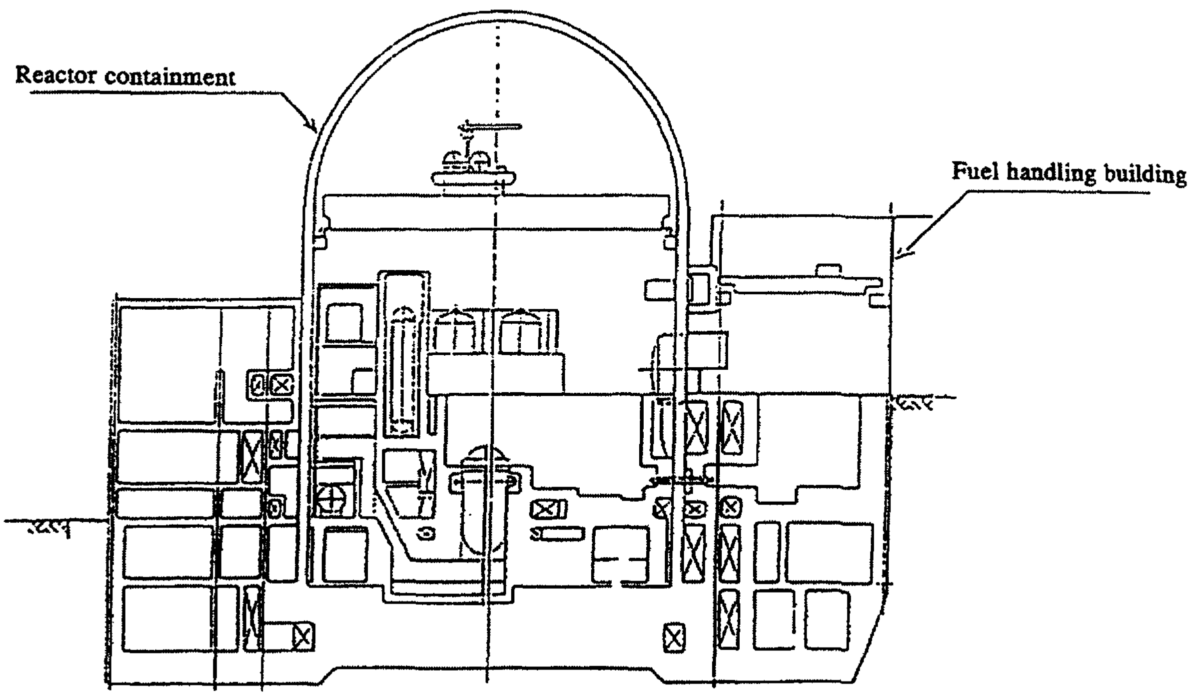

FIG. 3. APWR – Reactor building.

- For severe accident mitigation, countermeasures such as the use of the containment vessel air re-circulation system, alternate sprays supplied from the fire service water system, countermeasures for hydrogen control, water injection into the cavity from the fire service system, improvements of the cavity shape are adopted.

An enclosed space (annulus) is formed around the lower part of the containment shell to provide a double containment considering the possible release through the containment penetrations. The design features of the containment system are described in Table 2.

4.3. EP 1000, Westinghouse, USA / GENESI, Italy

In 1994, a group of European utilities initiated a program designated EPP (European Passive Plant) to evaluate Westinghouse passive nuclear plant technology for application in Europe. Based upon the extensive analysis and testing efforts for the AP-600 and SPWR passive plants, the EP 1000 uses passive safety systems to further enhance plant safety as described in Table 3.

TABLE 2. APWR CONTAINMENT SYSTEM DESIGN FEATURES

	Design Feature	Purpose/Description
Containment	Large dry cylindrical steel containment	Dimension (diameter/height): 45.5/69 m Free volume*: m^3 Design pressure/temperature: ~390/144 kPa/°C (severe accidents) *: kPa/°C
	Secondary containment provided?	Yes
Containment Heat Removal	Containment spray system	Containment heat removal. 4 sub-system (4 spray/residual heat removal pumps and coolers).
	IRWST	Eliminates the need of switch-over from injection to recirculation mode and provides cooling water to reactor cavity.
	Containment heat removal backup system	For unavailable containment spray system, provide normal containment vessel air recirculation system and alternative containment vessel spray supplied from fire service water system
Hydrogen Control	Purge system Recombiner (if necessary)	Hydrogen control
	Igniters	Prevents hydrogen detonation
Fission products Control	Containment spray system and Containment heat removal backup system	Fission product removal in containment during DBA or beyond DBA
Containment Isolation and leakage rates	Containment isolation system	Minimize containment penetration
	Containment leakage rates	0.1 vol %/day
Containment Overpressure Protection	Containment overpressure protection system	The system that provides for C/V heat removal backup system using normal C/V air recirculation system and C/V spray supplied from fire service water system
RCS Depressurization	Automatic depressurization system	Depressurize reactor pressure vessel and prevents high pressure core melt. Minimize probability of DCH
Reactor Cavity Configuration	1 m thick protective layer	Prevents core-concrete interaction
	Reactor cavity floor	Provides sufficient floor area for cooling of molten debris
	Labyrinth design	Retains the core debris and minimizes the potential of DCH
	Cavity flooding system	Cavity flooding from fire service water system for cooling the core debris and scrubbing fission products release

* Parameters that are not known or will be determined

TABLE 3. EP 1000 CONTAINMENT SYSTEM DESIGN FEATURES

	Design Feature	Purpose/Description
Containment	Dry, steel cylindrical containment Secondary containment provided?	Dimension (diameter/height)*: 46/67.8 m Free volume*: 82,000 m^3 Design pressure/temperature: 344/140 kPa/°C (severe accidents)*: kPa/°C Yes (partial)
Containment Heat Removal	Passive containment Cooling system (PCCS) IRWST	Provides containment cooling with passive water spray and air cooling, and keeps pressure within design limit Provides long-term injection water by gravity and provides cooling water to reactor cavity
Hydrogen Control	Passive autocatalytic Recombiner (PAR) Igniters	DBA hydrogen control Prevent hydrogen deflagrations or detonations.
Fission products Control	Natural process IRWST	Remove airborne activity by sedimentation and deposition. Fission product scrubbing during feed and bleed operation
Containment Isolation and leakage rates	Containment isolation System Containment leakage rates	Reduction of normally open containment penetrations by 60 % and no penetration required to support post-accident mitigation function 0.12 vol %/day
Containment Overpressure Protection	Containment with passive cooling	Containment pressure stays well below the predicted containment failure pressure with only passive air cooling.
RCS Depressurization	Automatic depressurization System	Depressurize RCS through PZR relief valves and prevents high-pressure core melt. Minimize the probability of DCH
Reactor Cavity Configuration	In-vessel retention(IVR) of core debris	Provides external vessel surface cooling with water from RCS break or IRWST water draining. Significantly reduce containment failure and radioactive release due to ex-vessel severe accident phenomena.

* Parameters that are not known or will be determined

Two containment designs are considered for the EP 1000 – single steel containment (Figure 4) and double concrete containment (Figure 5). For single steel containment, the steel vessel itself provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. During an accident, the air-cooling is supplemented by evaporation of water. In the case of double concrete containment, intermediate loop and external heat exchangers designed to maintain pressure boundary integrity also in case of mechanical failure of an internal heat exchanger.

The SSE specified in the EUR is 0.25 g, which applies to the standard design.

4.4. EPR, NPI, France/Germany

The EPR (European Pressurized Water Reactor) is being developed by Nuclear Power International and its parent companies, Framatome and Siemens, in cooperation with EdF and German utilities. The strategy pursued is further enhance the already high safety level attained at French and German plants. This strategy implies improving the prevention of accidents, including severe accidents, and adding features, mainly related to the containment, to mitigate the consequences of the postulated severe accident scenarios as described in Table 4.

The EPR is designed with double concrete containments; the inner containment of pre-stressed concrete and outer containment of reinforced concrete as shown on Figure 6. The leak-tightness requirement of less than 1 % volume per day is ensured without provision of a steel containment liner. Leakages through the inner containment are released via the annulus air extraction system.

The severe accident conditions lead to more severe design conditions compared to the existing plants. The most important factor is the increased design pressure, which was defined as 650 kPa (6.5 bar abs.).

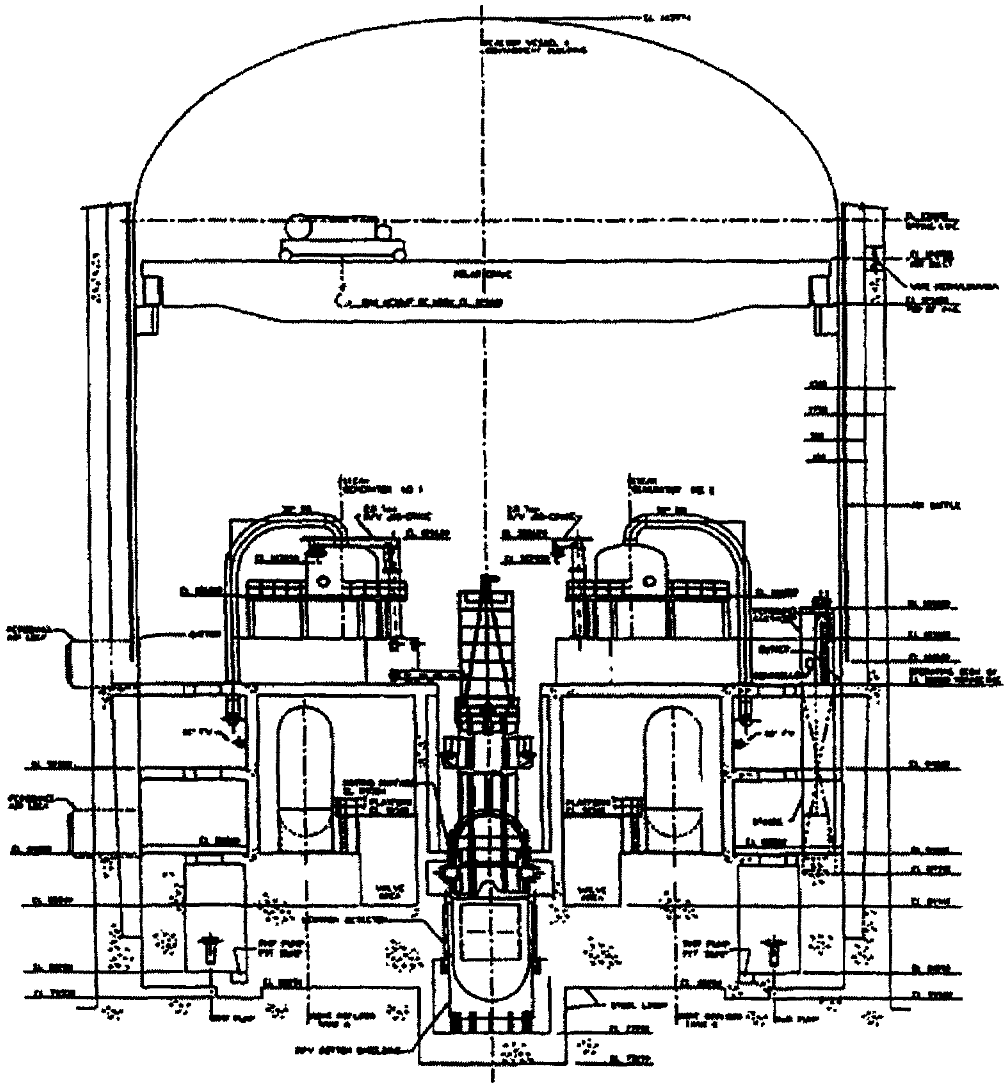

FIG. 4. EP 1000 – single steel containment

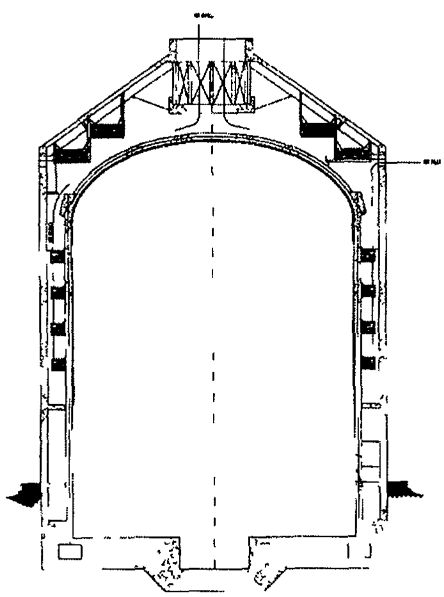

FIG. 5. EP 1000 – double concrete containment.

TABLE 4. EPR CONTAINMENT SYSTEM DESIGN FEATURES

	Design Feature	Purpose/Description
Containment	Cylindrical pre-stressed concrete Primary containment without steel liner Secondary containment provided?	Dimension (diameter/height)*: 48/ m Free volume*: 90,000 m^3 Design pressure/temperature*: 650/ kPa/°C (severe accidents) *: 650/ kPa/°C Yes
Containment Heat Removal	Containment heat removal IRWST Containment spray backup system	No active containment heat removal during DBA Eliminates the need of switch-over from injection to recirculation mode and provides cooling water to reactor cavity Provide containment heat removal in case of severe accidents
Hydrogen Control	Catalytic hydrogen Recombiner Catalytic hydrogen Recombiner (if necessary, Igniters)	DBA hydrogen control Highly reliable passive or active means for minimizing the threat of containment failure from hydrogen deflagrations or detonations.
Fission products Control	Containment spray backup system IRWST Annulus air extraction System	Fission product removal in containment during severe accidents Fission product scrubbing during feed and bleed operation Collection of unavoidable containment leakage in the annulus atmosphere and release via the stack after filtration. Minimizes offsite dose
Containment Isolation and leakage rates	Containment isolation system Containment leakage rates	Minimize containment penetrations 1.0 vol. %/day
Containment Overpressure Protection	Larger Containment	Larger containment prevents early containment failure or bypass. Ultimate pressure resistance to cope with energetic events.
RCS Depressurization	RCS depressurization System	Depressurize RCS through PZR relief valves and prevents high-pressure core melt. Minimize the probability of DCH
Reactor Cavity Configuration	Corium catcher Reactor cavity floor Cavity flooding system	Prevents base mat melt-through Minimizes non-condensable gas from core-concrete interaction Provides sufficient floor area for cooling of molten debris. Prevents ex-vessel steam explosion by minimizing the amount of water. Cavity flooding from IRWST for cooling the core debris and scrubbing fission products release

• Parameters that are not known or will be determined

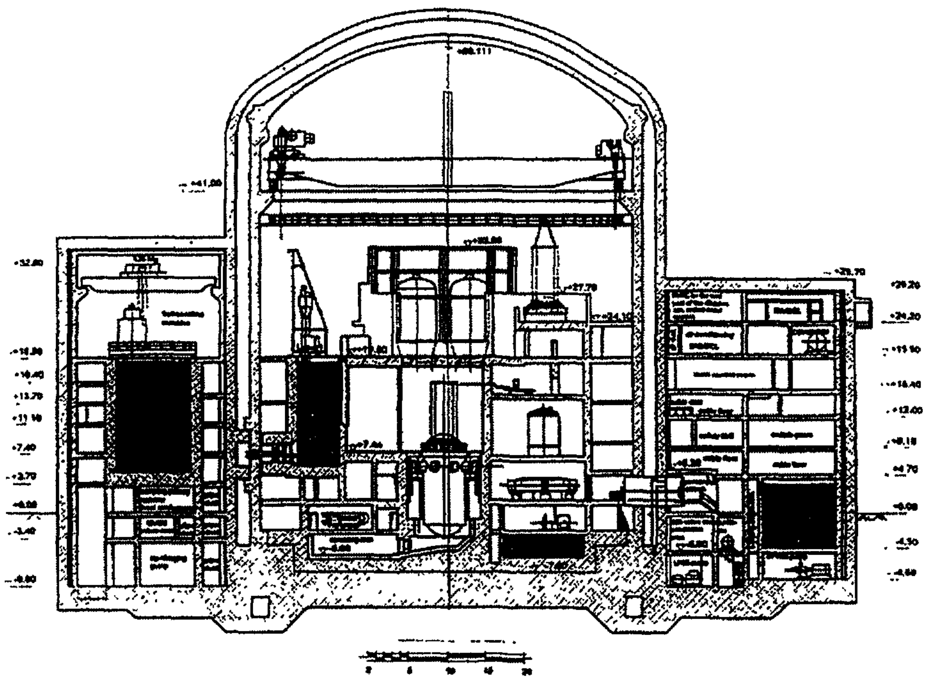

FIG. 6. EPR – containment building.

The plant is designed to withstand the impacts of the internal and external events. The seismic design is based on the spectrum defined in the EUR, scaled to 0.25 g, for the free field level of horizontal movement, for a wide range of soil conditions. For the explosion pressure wave, a maximum over-pressure of 10 kPa is the basis of the design.

4.5. ESBWR, GE, USA

The design of the ESBWR (European Simplified BWR) of General Electric represents a design for a 1190 MWe power plant, based on the earlier work done on the 670 MWe SBWR power plant.

The ESBWR containment structure as shown on Figure 7 is a reinforced concrete cylindrical structure, and is divided into a drywell region and a suppression chamber region with interconnecting vent system. Even though the design has not finished, external events such as SSE, aircraft crash, explosion pressure wave will be incorporated based on EUR requirements.

The basic safety design philosophy is on utilization of inherent margins such as larger volumes and water inventory to eliminate system challenges. The ESBWR design features are discussed in Table 5.

TABLE 5. ESBWR CONTAINMENT SYSTEM DESIGN FEATURES

	Design Feature	Purpose/Description
Containment	Concrete containment (drywell and pressure suppression wetwell)	Dimension (diameter/height) Lower drywell: 13.2/12 m Wetwell & upper drywell: 33/24.6 m Free volume*: m^3 Design pressure/temperature: 483/171 kPa/°C (severe accidents) *: kPa/°C
	Wetwell airspace size	Air space volume is sized for 100 % metal water reaction.
Containment Heat Removal	Suppression pool	Heat sink and pressure suppression for post-accident
	Passive containment cooling system (PCCS)	Provides long-term containment cooling and keeps pressure within design limit
Hydrogen Control	Inerting system	Prevents hydrogen detonation
	Passive autocatalytic Recombiner and igniters	Prevents hydrogen and/or oxygen combustion and detonation
Fission products Control	Suppression pool	Fission product retention for post-accident
	PCC Heat Exchanger	Filter aerosols – minimize offsite dose
Containment Isolation and leakage rates	Compact containment design with minimum penetration	Containment isolation with minimum leakage. High retention of aerosols.
	Containment leakage rates	0.5 vol %/day
Containment Overpressure Protection	Containment Overpressure Protection System	An optional system that provides additional defense in depth.
RCS Depressurization	Automatic depressurization system	Depressurize reactor pressure vessel and prevents high-pressure core melt. Minimize probability of DCH
Reactor Cavity Configuration	Core catcher	Retains molten core and prevents base mat erosion and melt through. Prevents core-concrete interaction
	Lower drywell floor	Provides sufficient spreading area (0.04 m^2/MWt)for cooling of molten core
	Flooding system	Provides additional cooling of corium on the floor.

* Parameters that are not known or will be determined

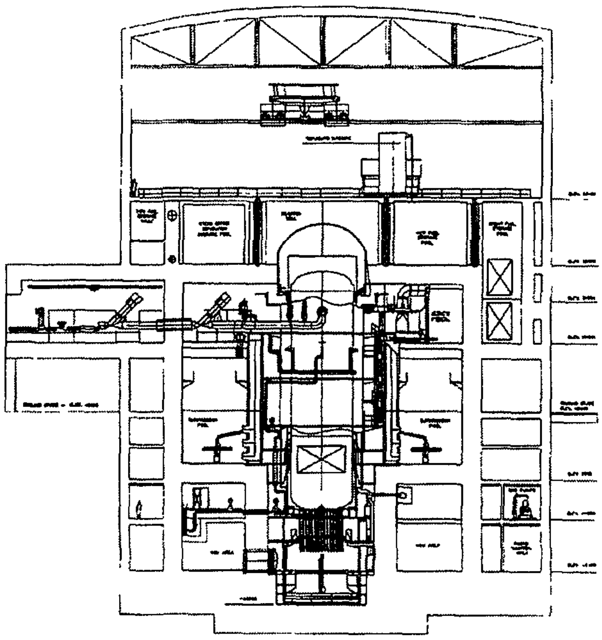

FIG. 7. ESBWR – reactor building arrangement.

4.6. KNGR, KEPCO, etc., Korea

The Korean Next Generation Reactor (KNGR) is a 1450 MWe PWR power plant, which has been developed for future use in Korea.

As shown on Figure 8, a double containment is provided consisting of an outer containment in reinforced concrete and an inner containment in pre-stressed concrete with a steel liner. The leak-tightness requirement of less than 0.5% volume per day is easily ensured due to the steel liner. Leakages through the inner containment are released via the annulus purge system.

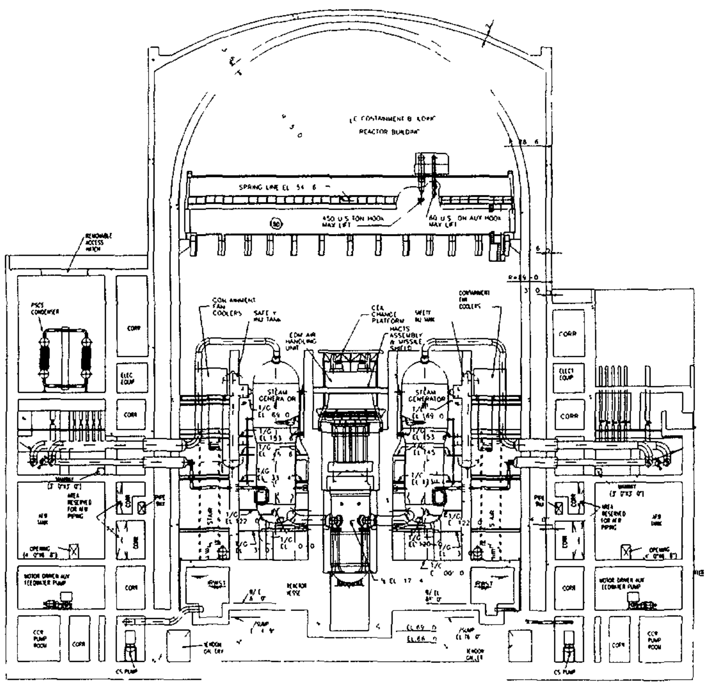

FIG. 8. KNGR – containment building.

Major changes in containment design compared with the Korean existing plants are:

- IRSWT is adopted, and the arrangement is made in such a way that the injected emergency cooling water can return to the IRWST.
- For severe accident mitigation, countermeasures such as reactor cavity design improvements, reactor cavity flooding system from the IRWST with active valves and passive fusible plugs, hydrogen igniter system are provided. Also, containment structure and reactor cavity will be designed to maintain the structural integrity during the postulated severe accidents.
- Double containment is selected, and leakages through the inner containment are released via the annulus purge system.
- The reactor cavity is sized and configured to spread out the ejected core debris over the floor surface area during a postulated severe accident, and is configured that steam exits the reactor cavity via a convoluted pathway that includes a 90 degree turn above the top of the core debris chamber virtually eliminating the potential for significant DCH-induced containment loadings.
- The diameter of the inner containment is increased from 43.9 to 45.7m, and the height is also increased to locate larger RCS arrangements, larger steam generators, a larger pressurizer, the IRWST, etc.

The integrated buildings comprising both containment and auxiliary buildings are analyzed for a SSE of 0.3g, and two control motions are applied to eight generic soil sites as well as rock site conditions. Design features are summarized in Table 6.

4.7. SYSTEM 80+, ABB CE, USA

The System 80+ has been developed by incorporating improvements into the System 80 design used for the plants in operation at the Palo Verde. Features that contribute to the safety improvements of the design include:

- Pressurizer volume is increased by 33%
- Secondary inventory in the steam generator is increased by 25%
- An IRWST acts as a quench tank for the SDS, avoids the need for safety injection re-circulation switch-over to the containment sump after LOCA, and provides a source of water for cavity flooding
- A large free containment volume provides additional margin against over-pressurization and ensures that global hydrogen concentration cannot reach detonable levels during an accident

The System 80+ containment is a 61 m diameter spherical steel shell with wall thickness of approximately 44 mm, and a secondary containment consisting of the containment shield building and the annulus as shown on Figure 9. The annulus ventilation system provides a mechanism for reducing unfiltered fission product releases following design basis and severe accidents.

The containment design pressure is 365 kPa and the design temperature is 143 ^{o}C. Calculations indicate that pressure limits determined in accordance with ASME Service Level C Criteria range from 895 kPa at a temperature of 143 ^{o}C to 826 kPa at a temperature of 232 ^{o}C.

Three control motions anchored to a 0.3g horizontal peak ground acceleration were developed and applied to twelve generic soil sites as well as rock site conditions.

The design features are summarized in Table 7.

TABLE 6. KNGR CONTAINMENT SYSTEM DESIGN FEATURES

	Design Feature	Purpose/Description
Containment	Cylindrical pre-stressed concrete of Primary containment (dry type) Secondary containment provided?	Dimension (diameter/height): 45.7/70 m Free volume: 90,441 m^3 Design pressure/temp.: 413/143.3 kPa/°C (severe accidents): ≤ ASME Service Level C Yes
Containment Heat Removal	Containment spray system IRWST Containment spray Emergency backup system	Containment mixing and heat removal. 4 train system (functionally interchangeable spray pumps and shutdown cooling pumps). Eliminates the need of switch-over from injection to recirculation mode and provides cooling water to reactor cavity Provide emergency backup by onsite pump from external water source for unavailable containment spray system,
Hydrogen Control	Passive autocatalytic Recombiner (PAR) PAR and igniters	DBA passive hydrogen control for containment and IRWST airspace. Highly reliable passive and active means for minimizing the threat of containment failure from hydrogen deflagrations or detonations
Fission products Control	Containment spray and emergency backup system IRWST Containment annulus purge system	Fission product removal in containment during DBA or beyond DBA Fission product scrubbing during feed and bleed operation Filter fission product leaked from primary containment. Minimizes offsite dose
Containment Isolation and leakage rates	Containment isolation system	Minimize containment penetrations
	Leakage rates	0.5 vol %/day
Containment Over-pressure Protection	Larger and robust Containment	Robust containment prevents early containment failure. Larger volume prevents late containment failure for more than 24 hours.
RCS Depressurization	Safety depressurization System	Depressurize RCS and prevents high-pressure core melt. Minimize the probability of DCH
Reactor Cavity Configuration	3 ft thick layer	Prevents base-mat melt-through due to core-concrete interaction
	Reactor cavity floor	Provides sufficient floor area (0.02 m^2/MWt) for cooling of molten debris
	Labyrinth design	Retains the core debris and minimizes the potential of DCH
	Cavity flooding system	Active and passive(fusible plug) cavity flooding from IRWST
Others	Passive secondary Condensing system (PSCS)	Provides passive decay heat removal through S/Gs using 2 condensers and condensing tanks from S/G to environment

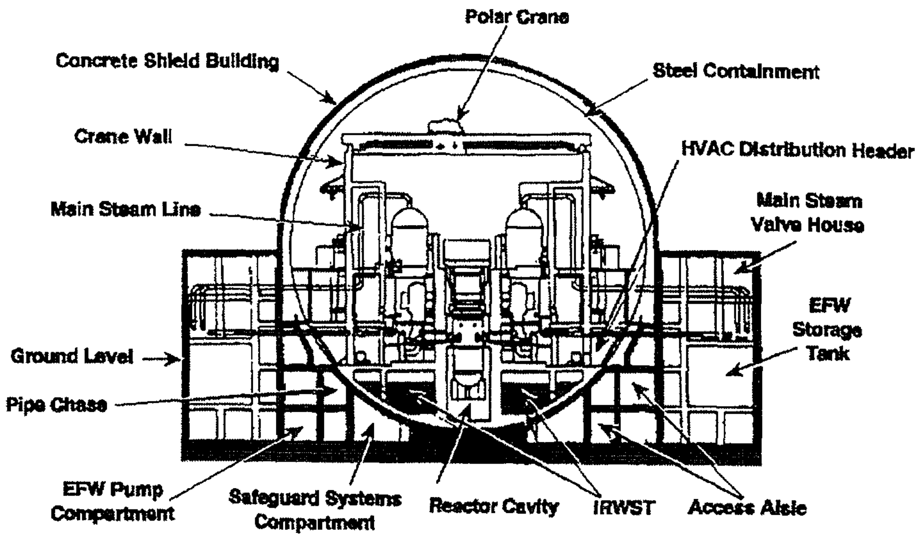

FIG. 9. SYSTEM 80+ - containment.

TABLE 7. SYSTEM 80+ CONTAINMENT SYSTEM DESIGN FEATURES

	Design Feature	Purpose/Description
Containment	Spherical steel of Primary containment (dry type)	Dimension (diameter/height: 61 m Free volume: 96,300 m^3 Design pressure/temperature: 365/143.3 kPa/°C (severe accidents): ≤ ASME Service Level C
	Secondary containment provided?	Yes
Containment Heat Removal	Containment spray system	Containment mixing and heat removal. 4 train system (functionally interchangeable spray pumps and shutdown cooling pumps).
	IRWST	Eliminates the need of switch-over from injection to recirculation mode and provides cooling water to reactor cavity
	Containment spray Emergency backup system	Provide emergency backup for unavailable containment spray system,
Hydrogen Control	Thermal Recombiner	DBA hydrogen control for containment and IRWST airspace.
	Igniters	Prevents hydrogen deflagrations or detonations
Fission products Control	Containment spray and emergency backup system	Fission product removal in containment during DBA or beyond DBA
	IRWST	Fission product scrubbing during feed and bleed operation
	Annulus ventilation system	Filter fission product leaked from primary containment. Minimizes offsite dose
Containment Isolation and leakage rates	Containment isolation system	Minimize containment penetrations
	Containment leakage rates	0.5 vol %/day
Containment Overpressure Protection	Larger and robust Containment	Robust containment prevents early containment failure (AICC load from 100 % metal-water reaction). Larger volume prevents late containment failure for more than 24 hours.
RCS Depressurization	Safety depressurization System	Depressurize RCS and prevents high-pressure core melt. Minimize the probability of DCH
Reactor Cavity Configuration	Limestone aggregate Concrete with 3 - 5 ft thick	Prevents basemat melt-through. Minimizes non-condensable gas from core-concrete interaction
	Reactor cavity floor	Provides sufficient floor area (0.02 m^2/MWt) for cooling of molten debris
	Labyrinth design	Retains the core debris and minimizes the potential of DCH
	Cavity flooding system	Active cavity flooding from IRWST for cooling the core debris and scrubbing fission products release

4.8. AP-600, Westinghouse

The AP-600 is a 600 MWe PWR with advanced passive safety features and extensive plant simplifications. The AP-600 is conservatively based on proven technology, but with an emphasis on safety features that rely on natural forces.

As shown on Figure 10, the AP-600 containment vessel is a cylindrical steel vessel. The containment vessel and the passive containment cooling system are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents. The shield building is also an integral part of the passive containment cooling system. The passive containment cooling system air baffle is located in the upper annulus area. The function of the air baffle is to provide a pathway for natural circulation of the cooling air. The passive core cooling system is also located in the containment building.

The AP-600 passive safety-related systems are summarized in Table 8.

Seismic design is based on the SSE of 0.3g.

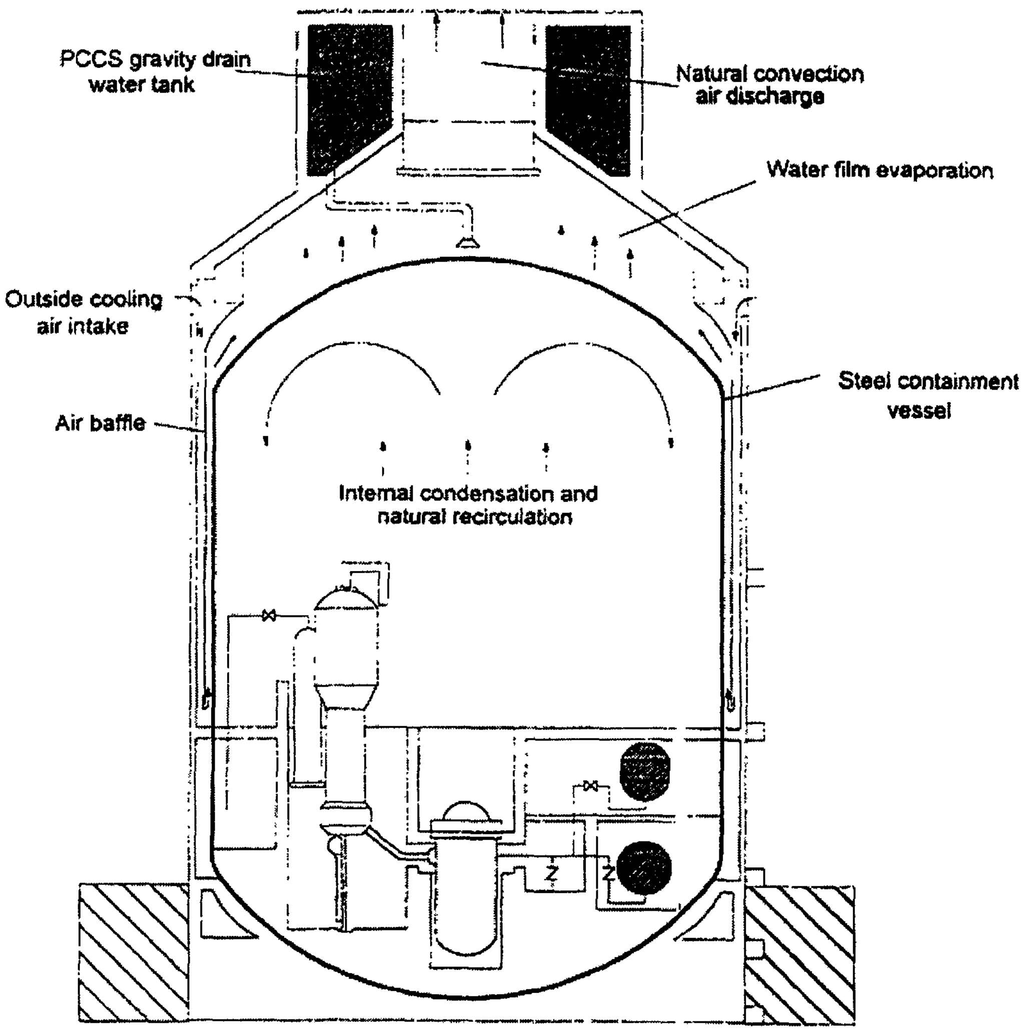

FIG. 10. AP-600 – containment vessel.

4.9. CANDU-9, AECL, Canada

CANDU 9 is a single unit adaptation of the Ontario Hydro Bruce B and Darlington plants 900 MWe class plants. The CANDU 9 containment is a steel-lined, pre-stressed concrete cylindrical structure with a hemispherical dome. The design leakage rate for the containment is 0.2% volume per day at the design pressure of 210 kPa(g). The containment is also designed for a main steam line break pressure of 450 kPa(g). Overall, the containment design has been developed to support an exclusion area boundary of 500 m.

In the CANDU 9 containment, the perimeter wall is separate from the internal structures as shown on Figure 11. This provides flexibility in construction, eliminates any interdependence between the containment wall and the internal structures, and provides additional flow paths for hydrogen mixing via natural circulation. Forced mixing of hydrogen gas is also achieved by means of large ducted fans that mix the accessible and inaccessible areas of containment post-accident, and large unducted fans in the dome area ensure a well mixed qualified, Class III electrical power. The fans are also associated with air coolers for long-term heat removal. Additional hydrogen mitigation measures include hydrogen igniters for short term and passive auto-catalytic recombiners for long term.

The design features are summarized in Table 9.

TABLE 8. AP-600 CONTAINMENT SYSTEM DESIGN FEATURES

	Design Feature	Purpose/Description
Containment	Cylindrical steel of Primary containment (dry type) Secondary containment provided?	Dimension (diameter/height): 39.6/57.6 m Free volume*: m^3 Design pressure/temperature: 316/137.8 kPa/°C (severe accidents): 316/137.8 kPa/°C Yes (partial)
Containment Heat Removal	Passive containment Cooling system (PCCS) IRWST	Provides containment cooling with passive water spray and air cooling, and keeps pressure within design limit Provides long-term injection water by gravity and provides cooling water to reactor cavity
Hydrogen Control	Passive autocatalytic Recombiner (PAR) Igniters	DBA hydrogen control Prevent hydrogen deflagrations or detonations.
Fission products Control	Natural process IRWST	Remove airborne activity by sedimentation and deposition Fission product scrubbing during feed and bleed operation
Containment Isolation and leakage rates	Containment isolation System Containment leakage rates	Reduction of normally open containment penetrations by 60 % and no penetration required to support post-accident mitigation function 0.12 vol %/day
Containment Overpressure Protection	Containment with passive cooling	Containment pressure stays well below the predicted containment failure pressure with only passive air-cooling.
RCS Depressurization	Automatic depressurization System	Depressurize RCS through PZR relief valves and prevents high-pressure core melt. Minimize the probability of DCH
Reactor Cavity Configuration	In-vessel retention (IVR) Of core debris	Provides external vessel surface cooling with water from RCS break or IRWST water draining. Significantly reduce containment failure and radioactive release due to ex-vessel severe accident phenomena.

* Parameters that are not known or will be determined

TABLE 9. CANDU-9 CONTAINMENT SYSTEM DESIGN FEATURES

	Design Feature	Purpose/Description
Containment	Prestressed concrete of a single containment	Dimension (diameter/height): 57/67.5 m Free volume: 124,000 m^3 Design pressure/temperature: 210/140 kPa/°C (severe accidents) *: kPa/°C
	Secondary containment provided?	No
Containment Heat Removal	Air cooling units	Provides containment heat removal
Hydrogen Control	Passive Autocatalytic Recombiner (PAR)	DBA hydrogen control
	Igniters	Prevent hydrogen deflagrations or detonations.
Fission products Control	Natural process	Remove airborne activity by condensation
	Air cooling units	Containment mixing
Containment Isolation and leakage rates	Containment isolation System	Closes open containment penetrations on a LOCA pressure and activity signal
	Containment leakage rates	0.2 vol %/day
Containment Overpressure Protection	No	No
RCS Depressurization	Main steam safety valves for steam generators	De-pressurize steam generator on a LOCA signal
Reactor Cavity Configuration	Steel calandria	To support heat transport pressure tubes and to remove decay heat from moderator and shield cooling system

* Parameters that are not known or will be determined

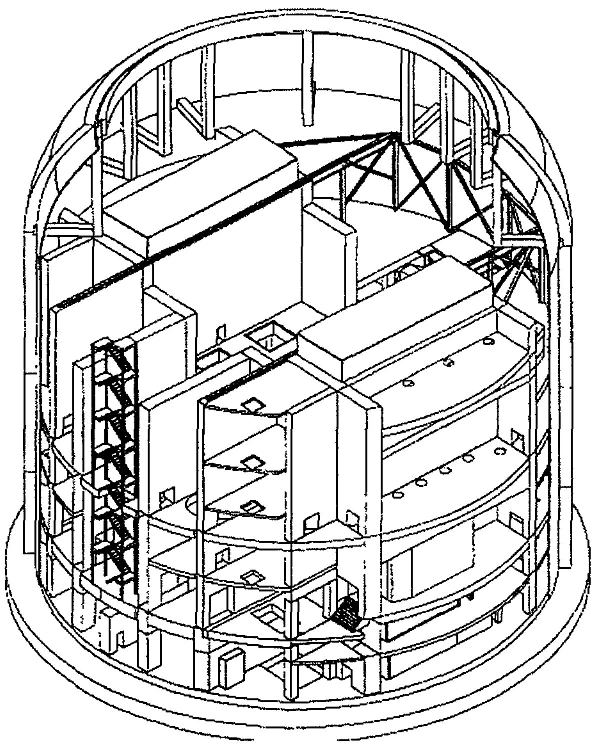

FIG. 11. CANDU 9 – containment.

5. CONCLUSION

Most of the proposed containment designs appear to be in compliance with the existing and newly emerging safety, technical and economic objectives, even though the approaches are slightly different from country to the plant. Concerns for severe accident protection and mitigation appear to be well integrated in the design, which are intended to provide a step forward in safety and technology.

The major advanced features adopted in the ALWR are as follows:

- large volume of robust containment
- double containment with an annulus filtered ventilation system
- a large reactor cavity for retention and cooling of core debris or a special design feature considering containment melt-through
- a reactor cavity flooding system
- adoption of IRWST for a PWR containment

Passive safety system has been applied for the medium size reactor, however, the concept may be extended to the large size units because of the inherent benefits utilizing the natural phenomena with design features simplifying the safety systems.

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BALANCING PASSIVE AND ACTIVE SYSTEMS FOR EVOLUTIONARY WATER COOLED REACTORS

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Abstract

Advanced concepts of the water-cooled reactors are intended to improve safety, economics and public perception of nuclear power. The potential inclusion of new passive means in addition or instead of traditional active systems is being considered by nuclear plant designers to reach these goals. With respect to plant safety, application of the passive means is mainly intended to simplify the safety systems and to improve their reliability, to mitigate the effect of human errors and equipment malfunction. However, some clear drawbacks and the limited experience and testing of passive systems may raise additional questions that have to be addressed in the design process for each advanced reactor. Therefore the plant designer should find a reasonable balance of active and passive means to effectively use their advantages and compensate their drawbacks. Some considerations that have to be taken into account when balancing active/passive means in advanced water-cooled reactors are discussed in this paper.

1. INTRODUCTION

The future of nuclear power depends essentially upon the two interconnected factors: how effectively and how safely it performs. The accumulated experience of nuclear power (around 450 commercial reactors with about 9000 reactor-years of operation at the end of this year) has demonstrated the good indices in both these aspects in comparison with the conventional power technologies. Few accidents have occurred in the history of nuclear energy, and the most serious ones turned out to be the results of the improper human actions or disconnection of those safety systems which have been designed specifically to prevent such accidents. So, many people's scepticism about the use of nuclear energy is not justified considering the causes and consequences of the events that have occurred.

To convince the public about the existing safety level and the competitiveness of nuclear power as well as to consider the increased demand of the public on safety, all organisations involved in nuclear power development and generation continue to give increasing attention to the safety and economics of the current and future nuclear power plants. In particular, great efforts are being devoted to this subject world-wide by designers, utilities and regulatory bodies as applied to advanced water-cooled reactor concepts. Many of these designs employ so-called passive features and means, some well proven by operation of existing reactors and others relatively novel. With respect to plant safety, application of passive systems/components is intended to simplify the safety systems and to improve their reliability, to mitigate the effect of human errors and equipment failures, and to provide increased time margin to enable the operators to cope with design basis accidents, as well as with design extension accidents.

The IAEA Conference on "The Safety of Nuclear Power: Strategies for the Future" [1] included discussions on the safety of future plants, and noted that "the use of passive safety features is a desirable method of achieving simplification and increasing the reliability of the performance of essential safety functions, and should be used wherever appropriate. However, a careful review of potential failure modes of passive components and systems should also be performed to identify possible new failure mechanisms". It was stressed that safety can be achieved by using either passive or active systems or a combination, and that both types of systems should be analysed from the standpoint of reliability and economics.

The application of passive means is connected with some problems which have to be solved by each plant designer. The passive systems have their own advantages and drawbacks in comparison with the active systems both in the area of plant safety and plant economics. Therefore a reasonable balance of traditional systems and new passive means is adopted in many future reactor concepts as the possible way to improve safety and public acceptability of nuclear power, and at the same time to keep nuclear power competitive with conventional power technologies. Some considerations which have to be taken into account when balancing active/passive means in the advanced reactors are discussed in this paper.

2. ACTIVE-PASSIVE CATEGORIZATION

Consideration of the operating nuclear power plants and the advanced concepts shows that safety systems cannot be simply classified only by two terms "active" and "passive". We can often find passive and active means in one safety system or even in its separate components. The traditional emergency core cooling system of the pressurized water reactor could be mentioned as an example of a safety system where the hydro-accumulators (passive element) and high/low pressure injection pumps (active element) are being used.

In several IAEA Technical Committee meetings the general definitions, descriptions and explanations of passive/active systems have been given. According to the IAEA definition, a system should be classified as passive if it consists of only passive components and structures or uses active components in a very limited way to initiate subsequent passive operation. Usually a system should be classified as passive if no external input is needed to perform its safety functions; otherwise a system is considered as active. The above definition of a passive system allows the use of instrumentation and the one-time repositioning of valves if adequate passive power supplies (e.g., batteries) are available.

One may also ascribe the "degree of passivity" to the system depending on the existence and necessity of the moving fluids, moving mechanical parts and external initiating signals; again the "degree of activity" may be ascribed to an active system depending upon the necessity of human actions and external inputs to initiate or to operate the system. Such a classification on the scale from fully passive to fully active may be useful for the system evaluation. For this classification, some items, either the inherent characteristics of the system or needed for the system to perform its function may be indicated (e.g. moving fluid, moving mechanical parts, input signal, etc.). Depending upon the number of the proposed items which are needed, a classification of seven categories was proposed in the framework of an IAEA study [2], shown in table I.

The system which has none of the active items is the system of Category 1 and it has the maximum of passive safety features (such a system could be considered as fully passive). An example of fully passive means would be cooling by radiation directly to the environment. Category 7 describes the features of a fully active safety system that can be characterised by the presence of all the above items;

the fire detection and fighting system may be an example of this category. In the IAEA-TECDOC-626, passive systems are described in 4 categories (A, B, C and D) which are the same as categories 1 to 4 mentioned in the table above. The above 7 categories are generally in agreement with the discussions that have been carried out during the IAEA technical committee meetings in Vasteras (1988) and Julich (1994).

Table I. Degrees of passivity

Cat.	Criteria						Examples
	moving fluid	moving parts	signal inputs	extern. power	human initiation	human interact.	
1	no	no	no	no	no	no	cooling by radiation
2	yes	no	no	no	no	no	cooling by free convection
3	yes	yes	no	no	no	no	check valves, accumulators
4	yes	yes	yes	no	no	no	passive heat removal system
5	yes	yes	yes	yes	no	no	ECCS of current PWRs
6	yes	yes	yes	yes	yes	no	boron injection
7	yes	yes	yes	yes	yes	yes	fire detection and fighting

Passive systems fall into the categories 1 to 4, and categories 5 to 7 are usually called active systems. In categories 2 – 4, the fluid (e.g. air, water) moves without external energy due to thermal-hydraulic conditions, whereas in categories 5 – 7, fluid movements are supported by pumps driven by external energy. In categories 3 and 4 mechanical movement occurs due to imbalances within the system (e.g. static pressure difference) or due to the forces directly exerted by the process (e.g. energy input into the closed reservoir of fluid). In categories 5 – 7, mechanical movement is supported by external energy. The systems in category 4 are initiated by components that rely on electronic, electro-mechanic, hydraulic or pneumatic logic.

3. CONSIDERATIONS TO GOVERN THE BALANCING PASSIVE/ACTIVE MEANS

The above classification does not mean that a more passive system should be automatically considered as more reliable with regard to the fulfilment of the designated safety function. These categories are intended to illustrate the concept of the spectrum from active to passive components Both passive and active systems/components have advantages and drawbacks; therefore, a case by case evaluation must be made, considering at first the fulfilment of the required safety function with sufficient reliability but also other aspects as e.g. the impact on plant operation, design simplicity and - last not least - costs. The best effect for the plant safety may be achieved with a reasonable combination of active and passive features to assure a certain safety function. Combined usage of active and passive safety means for the advanced reactors may allow to decrease the sensitivity of the safety functions to common cause failure, to increase the plant safety and at the same time to improve its economic performance.

The comprehensive effects of the balancing of passive/active safety means on the overall plant safety can be quantified through the use of probabilistic safety assessment methodology, yielding the values of the core damage frequency (CDF) and the large off-site radioactivity release frequency (LRF). Also, the effect of passive features in the system design may be quantified deterministically in terms of the maximum tolerable inaction time (MIT), during which the designated safety function is assured even in the absence of any actions performed by either operator or by active components. A low value of CDF is an indicator of the robustness of design, and investment protection. A low value of LRF is important for environment protection and public acceptance. A high value of MIT deterministically provides a measure of robustness in the plant design for dealing with any unforeseen situations of the equipment failures and operator errors It should be noted, that today the acceptable or desirable figures for CDF and LRF are directly included in the normative documentation of some countries. For example, according to Russian safety standards, these figures should be less than 10^{-5} and 10^{-7}, respectively; among other factors, this large difference between CDF and LRF results in the relatively wide usage of passive means in the advanced concepts of the Russian plants VVER-1000/V-392 and VVER-640/V-407.

From an academic view point, to find the optimum balance of active and passive means, it will be desirable to minimise CDF and LRF, and maximise MIT, to the extent possible under given constrains (e.g., plant capital and operational cost). However, for the advanced reactor concepts incorporating a number of relatively new design features and accident scenarios to be considered, it is difficult to accurately quantify many of the inputs needed for computation of CDF and LRF. This may result in a rather large uncertainty in the predicted values of these parameters. Therefore, with regard to these criteria an equally dependable conceptual decision about the active/passive features coupling may be achieved, more quickly and economically, on the basis of engineering judgement applied in a qualitative manner.

This approach can be translated in the form of the following considerations:

- application of passive features should reduce the number of components, and yield design simplification, so that the number and complexities of safety actions can be reduced;
- the passive means should be taken, to the extent possible, from similar ones having certain operational experience at power plants or elsewhere, so that the efforts needed to demonstrate the reliability and licensability are not too large;

Passive systems should be applied with high priority whenever such systems can provide one or more of the following benefits:

- elimination of need for the short-term operator actions during accidents being taken into account in the design;
- minimisation of dependence on off-site power, moving parts, and control system actions for normal operation as well as during design basis and beyond design basis accidents;
- reduction in capital, operation and maintenance costs due to design simplification.

Thus, the reasonable balance of the passive and active safety systems in the advanced reactor concepts is based on the detailed consideration of their advantages and disadvantages as applied to their effect on the overall plant safety and total cost. In general, one should point out the most essential advantages of the passive systems/means as follows:

- passive systems do not depend upon external energy supply
- passive features simplify the safety system configuration and reduce the number of equipment
- passive components may be more reliable than the active ones for their designated safety functions, but this should be carefully demonstrated over the expected range of conditions and considering possible degradation mechanisms
- passive systems decrease the possibility of human errors
- passive systems make the plant less sensitive to plant equipment malfunctions and erroneous operator actions.

The main drawbacks of passive safety systems include the lower driving forces and less possibility to alter the course of an accident if something undesirable happens (i.e., less operational flexibility). Due to low driving forces, the operation of these systems may be adversely affected by small variations in thermal-hydraulic conditions. Besides, the current computer codes are not sufficiently validated for the relevant conditions and phenomena (low pressure, low driving heads, effect of non-condensables, boron transport at low velocities, and the like). Therefore, separate effect and integral tests may be required for the code assessment and for demonstrating the safety performance of the passive systems being proposed in the design. The lower driving forces might also lead to quite large equipment, and this factor may reduce the cost savings projected from elimination or downsizing of active components. Besides, larger components may cause additional difficulties in seismic qualification on some plant sites, and this issue should be taken into account when evaluating the core damage and large release frequencies. In many cases, sufficient operating experience of the passive system/component under real plant conditions does not exist; so time-and money-consuming research and development works may be needed individually for each advanced reactor concept.

The design decisions with regard to the balancing active/passive features may also depend upon the functions assigned to the given system. In particular, the system having an important role in the mitigation of severe accident consequences which is located in potentially contaminated area (e.g., the part of the containment cooling system which is located inside the containment) could be designed as passive as reasonably achievable. This is because of the difficulty or even impossibility of access to such areas and because passive components may not require maintenance even during long term operation.

4. BALANCE OF PASSIVE/ACTIVE MEANS IN THE ADVANCED CONCEPTS

Safety features desired in future plants have been summarised by INSAG-5 in "The Safety of Nuclear Power" [3]. It notes that the Basic Safety Principles of INSAG-3 [4] remain valid and should become mandatory, and that beyond the safety principles of INSAG-3, but in extension of them, are further opportunities for improvement of safety on which new plant designs should begin to draw. They include several design approaches such as avoiding complexity, reducing dependence on early operator action, among others, and include specifically giving consideration in the design process to passive safety features. INSAG-5 further notes that though it may seem evident that passive systems are always safer, that may not be so in all cases. There may be safety disadvantages that would outweigh the gain. The superiority of the choice should be shown by demonstration or analysis.

Both novel and more or less proven passive systems and features are proposed in many advanced water-cooled reactor designs [5]. Some designs have only added a few passive components to the traditional systems. Some other designs make wide use of the passive systems/components to ensure or to back up a number of safety functions, including the basic ones: reactivity control, fuel cooling and confinement of radioactive substances. Many advanced water-cooled reactor concepts have implemented or considered different passive means to ensure these functions. In particular such functions as the containment heat removal, hydrogen management, core debris cool down and prevention of base-mat melt-through are probably among the most appropriate areas for passive systems usage. For example, the EPR concept with large power while preferring mainly active means for the prevention of core melt accidents also makes significant use of passive systems and components to ensure the confinement of radioactivity after such an accident. A brief review of the design decisions (implemented or being considered) to enhance the basic safety functions in the advanced reactor concepts is given below in this chapter.

4.1. Reactivity control

Traditional gravity-driven (in PWR and PHWR) or gas-pressure driven (in BWR) control rods is the main system to ensure reactor scram in currently operating reactors and in the advanced concepts. The traditional control rods system of PWRs is generally not effective enough to bring the reactor to a cold shutdown state. Therefore the reactivity control function is supported by chemical and volume control system and by emergency core cooling system injecting the highly borated water to the reactor. Although very good reliability records exist for scram excitation, some failures of the gravity-driven control rod insertion have been recognised. The failures occurred for different reasons such as loose parts in the primary circuit, broken fingers of rod clusters, deformation of guide tubes, deposition of impurities, fabrication defects, etc. In most cases, the effects of those failures were a reduction of insertion speed or an incomplete insertion. Besides, some failure modes have been considered which could prevent the reactor scram altogether, and therefore the designers need to analyse Anticipated Transient Without Scram events.

Taking into account the above deficiencies, some advanced concepts have implemented additional passive means to enhance the reactivity control function. The Russian advanced reactor concepts WWER-1000/W-392 and WWER-640/W-407 have an increased number of gravity-driven scram rods to maintain shutdown margin even in the absence of boron supply during the reactor cool down. Also, for the WWER-1000/W-392, a special rapid boron supply system has been designed and tested as a diverse system to the gravity-driven scram system for this reactor. A concentrated boron solution tank is connected to the suction and discharge pipes of each main coolant pump. The valves in the connecting pipes will automatically open if there is a demand for reactor trip but the reactor power after some time is higher than its value after scram. The concentrated boron solution is supplied to the reactor due to pressure difference between discharge and suction of the main coolant pump (pump head); even in case of loss of power the pump head during coastdown is sufficient to push out all the boron solution from the tank. The operability of the system has been confirmed by extensive experimental investigation using a scaled model.

All CANDU plants built in the last 20 years have a rapid gadolinium nitrate injection system that can shut the reactor down as quickly as the shutdown rod system. This injection system uses high-pressure helium to inject a gadolinium solution into the low pressure moderator. Instrumentation separate from the rod system and other safety systems but with equal capability to the rod system is used to open quick-acting, fail-open valves between the helium gas and the gadolinium solution.

A rapid emergency boration system is also implemented in the Sizewell B PWR for diverse reactor shutdown. It consists of four tanks of boric solution (3 m^3 of 7000 ppm concentration of boron in each tank), connected to each cold leg. The inertia of the main coolant pumps is sufficient for the system to fulfil its function. Functional tests were carried out, including mixing tests in case of the system failing on one of the four loops, and the results were used in the safety analysis.

4.2. Fuel cooling

The safety function "fuel cooling during transients and accidents" is ensured by provision of sufficient coolant inventory, by coolant injection, by sufficient heat transfer, by circulation of the coolant, and by provision of an ultimate heat sink. Depending on the type of transient or accident, a subset of these functions or all of them may be required. Various passive safety grade and safety relevant systems/components are proposed for future reactor concepts to fulfil these functions.

It is a feature of many advanced concepts that the water for replenishment of primary coolant inventory is entirely stored inside the containment. This ensures protection against external events and reduces the risk of loss of coolant accidents with containment bypass. Additional features implemented in some new designs to improve the replenishment of primary coolant inventory function include:

- pressurizer relief via the relief tank to the water storage tank;
- removal of heat from the primary circuit to the water storage tank via heat exchangers located in the water storage tank;
- water storage tank combined with the containment sump;
- water storage tank located at higher elevation than the reactor core for gravity-driven injection;
- storage of a portion of water at high elevation under the full primary pressure for coolant injection at high pressure.

Most of the new concepts suggest a combination of different passive and active means to ensure the function "coolant injection". Passive injection systems at high primary pressure are new in comparison to systems in operating reactors. AP-600 is an example of a design where this function is provided by core make-up tanks (CMT). Pneumatic isolation valves in the injection lines open automatically if one of the initiation setpoints (e.g. low primary pressure, low pressurizer level) is reached. These valves are fail-safe since they will open even if AC power fails. As long as the reactor coolant system (RCS) is still filled with liquid, cold water from the CMT flows to the RCS by natural recirculation. After the coolant starts to boil, steam enters CMT, the natural recirculation is terminated and injection to the RCS continues due to gravitation. To assure continued injection by medium and low pressure injection systems before the CMTs are empty, stepwise depressurization of the RCS is initiated if the liquid level in the CMT falls below defined setpoints.

Passive accumulator injection at medium primary pressure is applied in current pressurized water reactors as well as in the advanced concepts. Improvements of efficiency have been suggested for the future reactors on the basis of experience, such as optimised initial pressure, water/gas ratio, flow resistance in the injection line. Also, the abolition of the isolation valves in the injection lines is being considered in some new designs to increase the system reliability. The tendency in some advanced designs in comparison with the existing plants is to widen the primary pressure range for passive injection and to make it more controllable. The American AP-600, Russian W-392 and W-407, Mitsubishi APWR and Indian AHWR designs could be mentioned as examples of this tendency. In particular, Mitsubishi APWR designs make use of an advanced accumulator system to ensure the safety functions of core cooling. It has the function of both the accumulator tank and the low-pressure injection pump of conventional plants. So, the low-pressure injection pumps are eliminated and the safety injection system configuration is simplified.

Passive low pressure injection is foreseen in some new concept to replace or to back up the traditional pump injection being used for the operating plants. To ensure passive injection, the traditional water storage tank can be installed at higher level than the reactor core or special low pressure injection tanks at high elevation can be provided. Since the water level is at containment atmosphere, injection by gravity can only take place after complete de-pressurization of the reactor coolant system. This is accomplished e.g. by the last step of the de-pressurization sequence in the AP-600 design or by the special de-pressurisation system in the WWER-640/W-407 design; this system starts passively when the primary pressure decreases below 6 bar

The function "provision of sufficient heat transfer" in the advanced concepts is ensured in the same fashion as in currently operated reactors. This function is assured as long as sufficient water is supplied to the fuel rods. Sufficient water in the core is provided by the systems ensuring injection of the coolant as described above. Heat transport in reactor designs using mainly passive means is ensured during accidents by natural circulation between the core as heat source and heat sink (e.g. steam generators as in the Russian WWER-1000/W-392 design or heat exchangers in the water storage tank as in AP-600 design); the natural circulation may exist in single phase, two-phase and boiler-condenser modes. Some advanced designs make use of relatively new natural circulation paths, e.g. natural circulation after LOCA between sump and core via the sump screen and broken pipe in AP-600 or between the core, the flooded pool around the reactor and the spent fuel pool via the depressurization pipes and further connection pipes in WWER-640/W-407 design. The Indian AHWR uses natural circulation driven core heat removal during normal operation and hot shut down, making the core heat removal capability immune to the station black-out event.

The function "ultimate heat sink" for accident conditions in the advanced concepts is mainly ensured either by the water stored in tanks (located inside or outside the containment) or by heat transfer directly to the surrounding atmosphere (via special heat exchanger or via containment shell). In the first case, the heat sink may be limited in time, and human actions are required to restore it. For this type of the ultimate heat sink, the passive containment cooling water storage tank in the AP-600, which is needed especially for accidents in the design extension area, or the water tanks for passive containment cooling and for passive decay heat removal in W-407 and AHWR designs are examples. An example of the unlimited heat sink is the use of air heat exchangers in WWER-1000/W-392 design located outside the containment.

Another aspect of heat sinks that is sometimes made passive is the feedwater to the boilers. In CANDU, for example, there is gravity feed from an elevated tank into the boilers. High capacity valves can be opened in the steam system to depressurize the boilers and allow gravity flow for makeup.

4.3. Confinement of radioactive substances

This safety function is ensured by protecting and maintaining the integrity of the potential radioactivity release barriers (fuel, reactor system boundary and containment). These barriers are passive components as themselves; in addition, several passive means are proposed in new reactor concepts for the protection of these barriers. Most of these means are derived from design backfitting programs of existing plants, others are relatively novel. As far as the fuel and the pressure boundary are concerned, most of the considerations are the same as for the existing plants. New applications are mainly in the area of containment protection.

Essential reduction of radioactive releases intended for the advanced plants implies a significant reduction of the probability of core degradation up to core melt and - concerning the last level of defence-in-depth - a significant reduction of potential sources for radioactive releases in core melt scenarios. Severe accidents are considered already at the design stage of new concepts, so that the associated maximum conceivable release would necessitate only very limited off-site protective measures in area and time. The advanced concepts imply substantial improvement of the containment functions with respect to the radioactivity confinement in case of a core melt accident, and passive systems play an important role to achieve this objective.

Containment over-pressurisation may be avoided by passive containment cooling (e.g. this is proposed in AP-600, AHWR and W-407 designs) or by spray systems with preferably passive components inside the containment. Such design requirements are derived from the conditions inside the containment resulting from active systems failures. It also has to be considered that maintenance of the systems inside the containment may be impossible because the containment is not accessible during the long term phase of an accident. Filtered venting systems (like those at some existing plants) are also being considered in some concepts (e.g. WWER-1000/W-392) to prevent containment over-pressurisation. These systems are to be designed to follow the current long-term requirements in this area (e.g. filtered venting should not increase the risk of losing the containment function, filtered venting is not required in the short term of a core melt accident up to 24 hours, etc.). Some designs (e.g. AHWR) are being developed to incorporate passive means for isolation of the containment, using a water seal that gets established when a particular value of containment pressure is reached.

Special systems and components are being considered in many advanced concepts to solve the hydrogen-related problems during severe accident scenarios which are being considered in many advanced concepts. As a requirement, the containment volume should be designed large and strong enough to withstand a global deflagration of the maximum amount of hydrogen that can be contained in the containment atmosphere and also should resist a representative rapid local deflagration. Additional provisions are taken with respect to local detonation and to deflagration-to-detonation transition (DDT) sequences that might jeopardise the containment or its internal structures. A proper design of internal structures, catalytic devices for passive recombination of hydrogen, inerting of the containment atmosphere and other measures are taken to avoid dangerous concentrations of combustible gases (e.g. see GPR/RSK recommendations of 1993).

Penetration of the containment base-mat by molten corium must be avoided because this could result in a significant release and contamination of underground water and sub-soil. Passive core melt catching devices or specific spreading areas are suggested for this task in different advanced designs. Specific attention has to be paid for long term heat removal from the containment. For example in the

EPR concept, pipes are applied to connect the spreading compartment with in-containment water storage tank; these pipes are plugged by a fusible material. The plugs would be melted by contact with the corium, thus allowing the water to cool the corium in spreading compartment from the top.

5. CONCLUSIONS

The utilisation of passive systems in a reasonable combination with or instead of traditional active systems is being considered as an important measure to enhance the safety in many concepts of the next generation plants. Passive means have always been applied to reactor designs, and their wider usage in the advanced concepts is an available engineering option, not a safety objective by itself. Consequently no preference in general should be given for the use of either active or passive systems, and an individual evaluation is needed for each advanced design. The main criterion for the design decision is that the proposed system fulfils the required function to the appropriate reliability taking into account the existing constrains (e.g., for the plant economics).

The right balance of active and passive systems can be found only for each advanced concept separately, but the basic criteria for decision-making are the same for the most of the concepts. These criteria are mainly based on the weighing of passive and active system's advantages and disadvantages with regard to the designated functions, overall plant safety and cost. Some specific aspects should be reviewed when balancing passive/active means, such as:

- principle of defence-in-depth (e.g. multi-barrier concept), requirements of redundancy,
- diversification, single failure criteria, common cause failure modes;
- new accident scenarios such as inadvertent operation or interactions of systems;
- inspectability, recurrent testing, in-service testing close to operational mode;
- sensitivity to human errors and equipment malfunction;
- need of research and development work to demonstrate system operability.

There are some aspects in this area which are very plant specific, e.g. the validation of passive systems for plant conditions, integration of passive features in the overall safety systems, in-service inspection of passive components, etc. These problems have to be addressed by each plant designer to propose the optimal combination of active and passive systems and components. Nevertheless, one can conclude that passive systems/components have clear potential advantages in some applications. This conclusion is particularly true for beyond-design-basis accidents, and the passive means (systems) are being designed in many advanced reactors for severe accident mitigation. The design basis for these passive means (systems) is to be established with account for probabilistic safety criteria.

The IAEA, with international co-operation to elaborate global trends, has documented broad objectives for the development of advanced nuclear plants [6]. With regard to enhancing safety, its TECDOC-682 states that the plant design should seek to take the maximum, feasible advantage of inherent safety characteristics, and efforts should be made to utilise passive safety systems to the extent that they can be shown to be as reliable and cost effective as active systems for the same function.

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PROGRESS IN INSTRUMENTATION AND CONTROL INCLUDING THE MAN-MACHINE INTERFACE

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Abstract

The paper discusses benefits and difficulties associated with the use of new digital instrumentation and control (I&C) systems for nuclear applications. The rapid development of information technology has not been used to the same extent in the nuclear industry as in conventional industries. The most important reason for this under-representation is a de-facto moratorium for construction of new plants. In old nuclear power plants (NPPs) the new technology is used in modernisation projects and valuable experience has been obtained. The licensing of programmable systems presents new challenges especially for safety systems where a very high integrity is required. The paper discusses various aspects related to the use of the new systems for nuclear applications, it gives references to ongoing work of international organisations and to research that is seen as an effort to solve problems related to implementation of the new systems for nuclear applications.

1. INTRODUCTION

The development in the fields of electronics, computers and software has been very rapid over the last two decades. New generations of equipment with improved performance have been introduced to the market at a very high rate. This development is also reflected in new and improved systems for instrumentation and control (I&C) in industrial plants as well as in power plants. The new systems take advantage of technological achievements to accommodate more sophisticated and efficient treatment of measurement and control signals, for high speed and reliability, but also for high flexibility and versatility.

The new technology has made its way into the major industries, including conventional power plants. The nuclear industry has been slow in its adoption, however, in spite of the advantages that the new technology can bring compared with systems currently installed at the operating nuclear power plants (NPPs). A likely reason for this under-representation is that only few new plants have been ordered during the eighties when the new I&C technology matured. Another reason is a lack of pressure to exchange the old systems with more modern ones. A third reason is the traditional conservatism within the nuclear industry with its calls for proven designs. The situation is now changing. Currently installed systems, which typically were designed in the late sixties and early seventies, are becoming obsolete and there is a need for functional upgrades. As a result modernisation efforts are underway in a large number of NPPs in the world.

For new reactor designs, which have been developed by various vendors, the use of the new technology is a rule rather than an exception. It may be assumed that new reactor projects will rely much more on utilisation of information technology than projects in the past. A vision for the future is that I&C design and implementation are integrated into a frame of plant information management in a plant life time perspective covering all aspects of instruments, cables, signal conditioning, control room, man-machine interfaces, control equipment, process computers and other real-time computers.

2. DEVELOPMENT WITHIN INFORMATION TECHNOLOGY

2.1. Hardware

The development in electronics and computers has been tremendous during the last twenty years. The so-called Mores law predicts that the power of an electronics chip will double in 18 months, and this has come true over the whole period; computer hardware of today is four orders of magnitude more powerful than twenty years ago. The increased capability has been accompanied by a similar development also in hardware reliability and costs. As a result, the use of computers has expanded tremendously. In the fifties, people thought that a handful computers would have enough computing power to solve all the computing problems of the world, but today computers are used everywhere, and even integrated in toys for children.

2.2. Software

Software has experienced a similar rapid development as the hardware. Some twenty years ago software systems could typically include a few thousands lines of code, while similar systems of today can contain millions of lines of code. Today, software systems are built in several layers and new programming techniques have been introduced in a pursuit for better productivity in software projects. They are often characterised by overruns both in time schedules and costs, and techniques proposed have however not provided a remedy to this. A typical difficulty experienced in software projects is illustrated by the statement that "whatever the size of memory, one would always need 50% more". New software issued typically contains bugs that often are only a nuisance, but sometimes also may have catastrophic impacts.

2.3. Networking

The development of information networks is closely connected to the development in hardware and software. From the early computers that were interconnected with 300 baud modem lines we are today speaking of communication at speeds of hundred megabits per second, and in telecommunication technologies even more impressive performance is common. The development of networking has made it feasible to use distributed information processing where several computers communicate over a network. Computer networks may also be based on a large variety of media such as cables, optical fibres and wireless transmission. Networks make it possible to introduce systems redundancy in a simple way, but the network itself can be vulnerable for failures.

2.4. Implications for the systems

Rapid development is not always beneficial. A rapid introduction of new solutions tend to make earlier solutions obsolete and backwards compatibility of new systems is often poor. The degree of standardisation has been small and present systems often rely on de-facto standards that have developed from a market position. Many vendors have deliberately made their systems closed, resulting in interfacing difficulties between different systems. Spare parts for the systems are available only for a relatively short time. The driving mechanism for modernising of computer systems is quite often not the functions themselves, but increasing problems in getting spares for obsolete computers. When the price for a new component in the old computer exceeds that of a new computer, an exchange becomes quite natural. The obsolescence has also a bearing to people since it may be difficult for an expert on one type of systems to be in a similar position only a few years later. Similarly it may be very difficult to find experts capable of fixing problems in the old systems.

2.5. Management of software design projects

The management of software design projects has become increasingly difficult with an increasing complexity of the systems. The project management methods created to remedy these problems have more been based on sound engineering practices and quality control, than on the use of specific tools. One important approach is to establish detailed specifications of the final system before starting the design. A development in modular steps, with a detailed testing in each step, is another important component of successful design projects. A careful integration of the modules, and testing of the whole system before its final release, makes it possible to avoid many problems in later phases of the lifetime of the software. After the release of a software system, a systematic collection of problem reports and modification management will be important. Documentation has an important role in all phases of a software design project.

3. IMPLICATIONS FOR I&C SYSTEMS

3.1. A transfer from analogue to digital

The use of computers for I&C systems has introduced a transfer from analogue to digital signal representations and the use of sampled data systems in the control loops. These changes have brought along new design requirements, which to some extent have been put out of sight for the designers by the introduction of application programming languages. Compared with the mainly analogue systems, which were installed in the sixties and seventies, the new systems bring a number of benefits. The functions of the analogue systems were limited by both practical and economic constraints, while the new systems are far more flexible. Furthermore, the new systems have no drift, and signal storage capacity is not limited by physical restrictions. It is also easy to reach a very high accuracy in various steps of computations, and signals do not need scaling. It is easy to duplicate signals between various applications, and the complexity of calculations is no hindrance in building them. A better functionality of a control room can be achieved by means of visual display units (VDU). Digital systems are more reliable than analogue systems, require less maintenance and they usually have a longer expected lifetime. Back-up functions can be built both on a component and system level making the solutions fault tolerant. Advanced diagnostics and self-checking features are also easily included.

3.2. Typical I&C systems

Digital I&C system typically consists of the following major components [1]:

- hardware,
- systems software,
- applications software,
- process interfaces,
- a communication network
- man-machine interfaces.

The hardware can be configured in many different ways to yield solutions that are both efficient and reliable. The software is often divided into systems and application software to make it easy to configure for different applications. The process interfaces contain analogue and binary inputs and outputs, and also specialised interfaces to process components such as pumps and valves. The communication network is used to exchange information between various nodes in the system. The man-machine interface is often arranged through specialised nodes by which various displays and control panels can be connected to the system. Controls for various purposes such as interlocks, automatics and control loops are sometimes located to separate units connected to the communication network, but more often they are integrated in nodes driving the process interface. Higher level control is sometimes integrated in the communications or man-machine interface nodes.

3.3. Complexity and unpredictability

Digital I&C systems are more unpredictable than analogue systems. This unpredictability is due to the complexity of the software and to the fact that a small change in inputs may result in a very large change in the outputs of the system. In practice, this implies that it is not possible to use continuity arguments to predict in which range a certain output will be. The only way to predict the path that will be taken through some piece of software is to run it and observe the results. In practice it is very seldom possible to execute all possible paths of a certain piece of software, since these paths simply are too numerous. Unpredictability is also introduced through the reliance on software tools such as assemblers, compilers, linkers, loaders, operating systems, etc. that also may contain various errors.

3.4. New functions included

The possibility of building complex functions in software-based systems has been used in the modern digital I&C systems. Signals are represented directly in their engineering units, information can be stored for trend displays, advanced control algorithms can be utilised, alarm systems can include filtering, and various operator support systems can be built [2, 3]. Information can be duplicated to provide specialised displays for systems and plant states. It is possible to take an integrated approach towards information management and provide various personnel groups with the information they need. There are also possibilities to utilise new technology such as artificial intelligence, neural nets and fuzzy logic in an imitation of human reasoning.

3.5. Embedded systems

Another trend within I&C is to use computers embedded in various components. Various smart transmitters and intelligent valves and pumps are already used in the conventional industry, controlled by small computers embedded in the interface to the process component. The components can be connected to computer-based data concentrators through local communication buses and can be interfaced to local controls. Many systems such as access control, fire protection and ventilation system, which typically have not been a part of the I&C systems are now computerised and can easily be interfaced to the rest of the I&C. Computers are also used in stand-alone instruments used for various purposes. This provides a possibility to create special purpose interfaces, e.g. for communication between a calibrator and the calibrated component in exchanging messages of a successful calibration procedure.

3.6. Environmental compatibility

One specific concern related to new I&C systems is the environmental compatibility. The concern is raised through two mechanisms: on one hand modern electronic circuits are more sensitive to various disturbances; on the other hand they are using higher frequencies which may cause electromagnetic interference. Modern electronics is also more sensitive to environmental stress factors such as temperature, moisture and radiation. A remedy is to design robustness and to shield the components properly from various environmental impacts.

3.7. Commercial-off-the-shelf-systems

The nuclear power industry cannot be self contained with respect to its I&C solutions. Even if special nuclear grade systems are designed they will rely on electronics and software originally designed for other domains. In fact there is a trend to rely more and more on so called commercial-off-the-shelf systems (COTS). One can even say that a situation has emerged where one group of vendors is specialising on an integrating role and others on supplying components to be integrated into the systems. This specialisation yields a possibility for various vendors to concentrate on their core business, but it requires efficient communication between them to ensure that system requirements are appropriately reflected in the design of the COTS systems.

4. I&C APPLICATIONS IN NUCLEAR POWER PLANTS

4.1. General considerations

I&C system functions play an very important role in the operation and safety of NPPs. Proper initiating of safety functions depend on correct signalling and activation of various safety systems. The normal operations control has the task of preventing the plant state from moving into unsafe conditions. Correct and timely actions by the operators rely on correct and well presented information in the control room. In NPP applications, a distinction is made between safety systems, safety-relevant systems and non-safety-classed systems, and this yields a typical division into safety I&C, plant control systems and plant information systems. The I&C for NPPs must be designed to meet the general safety principles such as defence-in-depth and the single failure criteria, and it must be possible to verify that the design criteria are fulfilled.

4.2. The new systems in NPP applications

The technological development in information technology and I&C will obviously influence also NPP projects. Until now this influence has been relatively minor. The most important reason for this is that very few new NPPs have been ordered during the last twenty years. Another reason is the safety requirements, which "prescribe" proven technology. The pace of development with very short-lived product generations has made it very difficult to establish a technology that can be considered proven. Another problem relates to the complexity of software-based systems that makes it difficult to generate the required evidence that the systems will perform correctly in all possible situations. One specific difficulty is that many of the I&C solutions have been created for the conventional industry, and the nuclear industry has had no or very little opportunity to bring in its own special concerns and requirements into the design process. One way for vendors to overcome this problem is to have two base system variations in which a subset of the software has been validated more extensively to meet requirements of a nuclear-grade system.

4.3. Modernisation projects

Operating NPPs are facing an increasing obsolescence of I&C systems and, at the same time, requirements for improved competitiveness and safety [4]. Plant modernisation is a response to these changes in the environment. For the I&C systems, this covers a wide spectrum of approaches and strategies, ranging from ad hoc replacement of individual systems or functions to complete replacement, which in turn spans from one-to-one replacements, through an upgrading of old systems to an implementation of completely new systems. Experience from such projects e.g. in Finland, Germany, the Netherlands and Sweden shows that it is of paramount importance to establish a strategy for the remaining lifetime of the plant. A choice has to be made between a gradual replacement of the old systems over a series of normal outages or a single extended shut down for a complete replacement of the old systems. Regardless of the selected strategy one has to plan for a certain co-existence of old and new systems. Modernisation projects may also require a regeneration of the plant design base in which new safety requirements should be reflected [5]. Implementation of new I&C systems may be attractive for new plants of a standard series. An example is the upgrades of the Korea Standard Nuclear Plant (KSNP) design for the Ulchin NPP Units 5 & 6 under construction. These upgrades involve introduction of new NSSS Control system duplication, Plant monitoring system, Digital plant protection system and Digital engineered safety features actuation system, and represent pilot cases with respect to licensing of such systems in Korea.

4.4. Two paths of systems development

Development of I&C systems for nuclear applications follows two paths. One option is to build one-of-a-kind system very much from scratch, but relying on available assemblers, compilers, linkers and loaders. The other is to build on a well established I&C platform and implement system functions using an available application programming language. The first solution offers a possibility to gear the quality assurance to the special requirements of the nuclear industry, while the second solution provides the opportunity of having a far larger database of actual experience with the system. The first solution has the drawback that the experience with the system is minor, and problems of applying quality assurance to all parts of the software still exist. A problem with the second solution is that it may be impossible to gather necessary data for creating evidence that the system is good enough for its intended use.

4.5. Common-mode failures

Common-mode failures represent intrinsic difficulties for a design that aims at defence-in-depth; a coupling between redundant system that makes a common-mode failure possible, implies that the single failure criterion is not fulfilled. Common-mode failures can be avoided only if systems are truly independent. The potential for common-mode failures is much higher in software-based I&C than in analogue systems. Experience has shown that independence between software design projects is not sufficient, since specifications may contain errors that penetrate to the final code. Diversity is no solution to this problem, because the same type of electronic chips, the same compilers and the same thinking may have been used in creating the diverse systems. It has also been shown that an extensive base of experience from some applications does not necessarily ensure the reliability of the system in another application.

4.6. Verification and validation

The process of verification and validation (V&V) becomes crucial for the final quality of software-based systems. The complexity of the software makes it impractical to carry out the V&V process based only on testing of the final product. Instead the V&V process has to include inspection and review also of intermediate results and the processes behind them. In practice, this means that the V&V process should follow and have a close interaction with the design process. V&V can be facilitated by various tools by which the software can be checked automatically. One specific way of supporting the V&V process is to use formal specifications.

4.7. The main control room

The main control room requires special considerations in the design of a new I&C system [6]. The design of a computer-based control room is quite different from a conventional control room. One special consideration is to provide operators with an overview of the plant through the restricted window of a VDU. The allocation of control functions between the I&C and the control room operators is closely related to the level of automation. A high level of automation may ensure better repeatability and speed in the actions, but may leave the operator with tasks without a clear structure. Necessary information should be found easily in all operating situations, and in that context the structuring of information is important. Experience shows that it is highly recommended that the control room design be based on a detailed task analysis. Human factors engineering is another important aspect; various guidelines for performing control room design reviews are available [7].

4.8. Integrated plant information management

An integration of plant information is seen as a benefit from the new I&C systems. The information is functionally collected to a large database through one gate. Before information is entered into the database, an extensive validation is made to ensure that the value of the signal is correct. If not, the signal is marked unreliable. Information from the plant database can then be used anywhere without restrictions to place or time. Interfaces will be arranged between the plant database and various applications. The configuration management can be supported by interfaces to design, work order and maintenance systems. Control room operators and maintenance personnel can be supported by providing interfaces to the document management system. The creation and maintaining of PSAs can be supported by interfaces to the configuration management and the maintenance systems, etc. The main difficulty in the creation of such an integrated plant data base is the establishment of standardised and open interfaces to which various vendors can interface their own systems.

4.9. Information management in a plant lifetime perspective

The problem of information management can be extended to the whole plant lifetime. An extensive use of computers during design and construction implies a computerised design base. This gives extended possibilities to use computerised tools in the V&V process. Reuse of specifications generated for one purpose as input for configuring software for another application represent interesting new possibilities. Code and document generation can also be assumed to be relatively standard applications in future NPP projects. Possibilities of remote diagnosing and fault finding are already in use today. If suitable efforts are made to find invariants in the design of I&C systems, it may also be possible to reuse the system specifications for other hard- and software platforms. Modern telecommunication provides an opportunity for co-operation between design teams separated by large geographical distances.

5. THE LICENSING PROCESS

5.1. Requirements for licensing

The requirements depend on the safety importance of the system to be licensed, or qualified. Requirements may be placed even on non-safety systems when they are interfaced to safety or safety-related systems. With respect to requirements applicable to I&C systems we can separate between deterministic, probabilistic and human factors requirements. Due to the complexity of an I&C design process it is necessary to define requirements on the final product, intermediate products and the processes used to generate these products. Deterministic requirements are usually placed on solutions with limited design complexity, solutions for fault tolerance, spare capacity and quality control processes. Probabilistic requirements are often established to ensure that assumptions made in the PSA are valid both with respect to sequence modelling and reliability estimates. Human factor requirements are formulated to ensure that operators will be able to understand and operate the systems and that the first rapid state changes after the onset of an accident are automated. Requirements are also placed on the process of generating evidence that the licensing requirements are fulfilled.

5.2. Phases in the licensing process

The phases in a licensing process depend on legislation and regulatory practices. Early interaction between utility and regulator can be helpful. Typically, system architecture and design principles set the stage. Already at this phase, the regulator may require a V&V plan with descriptions of major project milestones and the quality systems to be used in the project. An assessment of specifications is often the next step in process. When testing is initiated, it is usual to require a comprehensive test plan. This includes both factory and site acceptance testing, which often are carried out in the presence of a representative of the regulator. A modern software quality control system includes several design reviews to be carried out when certain stages in the design have been completed. Some of these reviews may be performed by independent reviewers to ensure that also difficult questions will be brought into the open. If the software development process relies on the use of various tools such as code and documentation generators it may be necessary to license them in a separate process.

5.3. Collection of evidence

Typical software quality assurance procedures monitor process compliance more than direct product quality. The structure and implementation of these procedures represent one component in the compilation of evidence. Various intermediate products can also be assessed and reviewed in the course of the design project. Inspections of specifications, documentation and code can provide evidence that the underlying processes have been producing required quality. Special V&V tools such as machine code disassemblers, automated tools for inspection of assembly programs, tools for static and dynamic analyses of the software, etc. may be used to get evidence that the coding has been performed according to standards. The completeness of test programs can be assessed using various tools for instance to investigate sensitivity to artificial errors in the code. Statistical testing, either with random test inputs or inputs mimicking a certain usage profile can be used to collect quantitative evidence for the reliability of the software. Operational experience can provide some evidence, but the problem is to prove that the usage profiles of two different applications are similar enough.

5.4. Conditions for acceptability

To make the licensing process transparent it is important to have the conditions for accepting or rejecting certain solutions documented in a clear way. Regulatory requirements are not stable over time, because new experience may bring in needs for stricter acceptability criteria. Still, it is necessary to maintain consistency in the regulatory approaches. The safety importance for functions and components are reflected in the safety classification, but this is usually too crude to give clear guidance on the acceptability. Deterministic requirements can be checked by inspections, but the probabilistic requirements are more difficult. One possibility is to anchor acceptability conditions to a plant specific PSA where a certain reliability is required. Statistical testing can be used to collect evidence at some reliability level, but it becomes impractical for systems with a very high reliability requirement. For such systems it may not be possible to provide reliability estimates without relying on expert judgement. It is often beneficial to model software errors using some method like the Failure Modes and Effects Analysis (FMEA). A controversial issue in this context is requirements for diversity, since it can be very hard to verify the actual degree of independence.

5.5. Experience from licensing processes

Experience from software licensing has been obtained in Canada from the Darlington plant, in France from the so-called SPIN system and the N4 plants, in the United States from several upgrades of plant protection systems and in the UK from Sizewell B. Based on this experience the four regulators in Canada, France, USA and UK came together and developed their consensus on what should be included in the licensing process [8]. In Germany, the Siemens Teleperm XS system has recently been licensed in an extensive process involving several parties [9].

5.6. The Finnish YVL guides

One example of new licensing requirements for I&C can be taken from Finland. According to the Finnish regulatory system, STUK issues detailed safety requirements. These requirements, the so-called YVL guides, govern the practical safety activities at the nuclear installations and the safety inspections carried out by STUK. The guides are updated regularly and presently some sixty plus guides are maintained. The guides are not mandatory, but represent a strong regulatory position. One of the guides, the YVL-5.5 on "Nuclear power plant automation systems and equipment" has recently gone through an extensive revision and the new document is now (11/98) almost finalised. The revision process was initiated by the need to issue detailed requirements on programmable automation systems. The main difficulty in writing the guide has been to find a proper balance in the details of the requirements. The guide should be consistent with other guides, but the burden of proof should not restrain a transfer from technically inferior solutions. The Finnish regulatory practice requires a pre-inspection of relevant documentation before a project is released for implementation. For programmable automation systems this review can be performed in two phases, where general design criteria and solutions in principle are covered in the first phase. In the second phase, detailed information on the selected systems and the design should be provided together with the V&V plan. Before the installation, STUK also reviews acceptance tests, inspects the installation and monitors the start-up of the systems at the plant.

6. NEW PLANT DESIGNS

6.1. *Development by reactor vendors in the world*

All reactor vendors have prepared themselves for a move to new I&C systems. Many vendors have been involved in plant modernisation projects that have given them experience in the utilisation of the new systems. For new reactor concepts, the development of I&C systems has been on a more generic level. This is quite natural since final solutions will depend on the availability of specific systems at the time of construction. The new reactor designs generally fall into two categories: designs of evolutionary type and designs requiring substantial development [10]. The evolutionary designs will, to a large extent be configured and laid out in the same way as their forerunners, while the "developmental" types may incorporate significant conceptual changes, e.g. aimed at eliminating safety hazards and improving safety performance. The I&C solutions for the two categories do not differ very much, however. They are based on the same kind of digital distributed I&C systems and the control room is suggested to be compact and based on VDUs. The only difference between the two reactor designs categories is that some simplifications with respect to redundancy and physical independence are suggested for the "developmental" designs.

6.2. Similarities and differences in I&C solutions

When the I&C solutions proposed by various reactor vendors for their designs are compared, the differences are quite small. All reactor vendors move towards distributed digital systems. The control rooms are based on computers and VDUs. The level of automation is relatively high. Most of the evolutionary reactor concepts rely on a 2/4 redundancy principle. The development work of the Korean industry on the Korean Next Generation Reactor (KNGR) based on the System 80+ of ABB Combustion Engineering can be taken as a typical example. The main control room is a compact work station design that implements the utility requirements of the EPRI URD, featuring three redundant operator consoles, a separate safety console, a Large Display Panel, and monitoring consoles for the supervisor and technical advisor of the shift. The man-machine interface is based on computerised operating procedures and soft controls, and the I&C design is a complete plant-wide integration of digital technology. The Plant Protection and Safety Component Control System are four-channel, programmable logic controller-based systems. Non-safety controls are implemented in a two-channel system using diverse processors; plant monitoring is also provided by two independent and diverse systems.

7. RESEARCH ACTIVITIES

7.1. National research activities

All nuclear countries have various research activities going on. The activities can be divided into two parts: the more or less public research, and research driven by the nuclear vendors. The way these activities are organised depends on the country. In the USA, U.S.NRC and EPRI are funding and co-ordinating much of the activities. In France, IPSN is performing a large amount of the regulatory research. In Finland, VTT carries out research projects for both the utilities and the national regulator.

7.2. A report on licensing of I&C systems

The U.S.NRC, the regulatory body in the USA, has experienced various problems in their approaches to the new systems. In an attempt to get outside guidance the National Research Council was asked to conduct a study on application of digital I&C technology to commercial NPP operations. The study was carried out in two phases in which the first identified important safety and reliability issues arising from the introduction of the new technology. In phase two, the committee was asked to identify criteria for review and acceptance of digital I&C technology both in retrofitted and new reactors. In areas lacking sufficient scientific basis, the committee was asked to suggest ways in which U.S.NRC could acquire the required information. The work of the committee resulted in a comprehensive report where many important issues were brought up and discussed [11].

7.3. The OECD Halden Reactor Project

The Halden Project is an undertaking of national organisations in 20 countries sponsoring a jointly financed programme under the auspices of OECD/NEA. Discussions are under way for enlarging the member circle. Collaborations with East-European countries in support of plant safety and reliability are also expanding. The programmes aim at generating key information for safety and licensing assessments on extended fuel utilisation, degradation of core materials and man-machine interactions research. The activities in the man-machine area are highly relevant for the I&C solutions and they include a new man-machine research laboratory, plant surveillance and operations systems, assessments of system quality and several projects on software verification and validation.

7.4. Research needs

Research needs can be divided into two areas: licensing of programmable systems and human factors. For important safety systems there is a need in probabilistic terms to go beyond a reliability of 10^{-3} per demand, and this requirement is very hard to attain [12]. The formal methods represents an important portion of the methods for V&V of software, and one important research task is to collect evidence on their efficiency. In the human factors area, human reliability, cognitive errors and teamwork are important subjects for research.

8. INTERNATIONAL CO-OPERATION

8.1. IAEA

In 1974, the IAEA launched the Nuclear Safety Standards (NUSS) programme for the purpose of establishing internationally agreed safety standards for nuclear power reactors. The resulting codes and guides were published in the IAEA Safety Series. Now, the hierarchical structure for the Safety Series publications is Fundamentals, Requirements, and Guides. The activities related to I&C are located within the departments of Nuclear Power and Nuclear Safety. The I&C activities of the Department of Nuclear Power are co-ordinated through the International Working Group on Nuclear Power Plant Control and Instrumentation (IWG-NPPCI). Recent activities of IWG-NPPCI include the preparation of technical documents, specialists meetings and co-ordinated research programmes. An activity to identify IAEA publications in need of updating to reflect the new I&C systems has been initiated.

8.2. OECD/NEA

The Nuclear Energy Agency (NEA) of OECD is involved in I&C issues through activities within the Committee on Safety of Nuclear Installations (CSNI) and the Committee on Nuclear Regulatory Activities (CNRA). An International Workshop on the Technical Issues of Computer-Based Systems Important to Safety was arranged in March 1996 in Munich [13]. This workshop included several presentations on the state-of-the-art in the licensing of software-based systems. Within CSNI the Principal Working Group #1, on Operating Experience and Human Factors, has established a Task Group that has the objective to establish and develop a database on operational experience with Computer-Based Systems Important to Safety in NPPs [14]. OECD/NEA has also in a Senior Group of Experts on Safety Research (SESAR) identified research strategies and needs and in this work among other issues addressed Plant Control and Monitoring and Human Factors which are relevant within the I&C field.

8.3. The European Union

During 1994-98, research and technical development activities have been carried out under the Fourth Framework Programme and the parallel Euratom framework programme that covers research and activities in the nuclear sector. The Programme is implemented through 18 specific thematic programmes grouped under four priority areas. The total budget for the Programme has been more than 13 billion ECUs. Within the programme various projects related to nuclear power and I&C have been financed. A continuation of earlier programmes is proposed in the Fifth Framework Programme. The Programme itself has not yet been adopted, but in-depth discussions have been started on specific themes to be implemented from 1999 onwards. Also this programme is expected to contain activities related to nuclear power and I&C. In addition to these research-oriented activities, the European Commission sponsors various working groups that take a stand on important issues. One of these the Nuclear Regulator Working Group (NRWG), has prepared a Document on Regulators' Current Requirements and Practices, which discusses licensing of programmable systems.

8.4. Standardisation organisations

Many organisations prepare standards relevant for I&C in NPPs. The most important of these international standardisation organisations are IEC and ISO. IEC, the International Electrotechnical Commission, is an organisation of 50 countries involved in standardisation in the fields of electricity, electronics and related technologies. ISO, the International Organization for Standardization is a world-wide federation of national standards bodies from some 100 countries writes other standards. Most international standards are written by working groups that comprise technical experts from different countries. The experts are appointed by countries, but serve as individuals, and the expert's action in the group is not necessarily reflecting national positions. The working group draft may be approved as a draft standard, and can after that be accepted as an international standard, through voting by the official delegates of the IEC and ISO member countries. In addition IEEE, the Institute of Electrical and Electronic Engineers, is another important standardisation organisation working mainly in the USA. Some important standards for I&C in NPPs are IEC-880, ISO 9000, ISO 9000-3, IEEE 730, IEEE 1012, and IEEE7-4.3.2-1993.

8.5. Utility requirements

Utilities both in Europe and the USA have initiated work aimed at creating common utility requirements for new plants to be built. One rationale in the work has been to establish a common approach that could ease the licensing process. Another benefit of the work process is that it may help harmonising of views on crucial safety issues.

9. CONCLUSIONS

In assessing the overall situation it is evident that distributed digital systems are the only realistic I&C alternative for both modernisation and new NPP projects. The licensing issue of the new systems has not been completely resolved, however. A resolution would imply estimating in some believable way the reliability of a system containing both hardware and software. Before this can be accomplished, further research efforts are needed.

At present, the majority of concrete projects are I&C modernisations in operating NPPs. A major difficulty in these projects arises from the design constraints given by the actual layout and process configuration; this means that the full benefit of the new information technology remains to be realised in an NPP project. Among the modernisation projects, some have been relatively successful and others less successful. Some of the difficulties go back to the problems described in this paper.

With respect to I&C, the nuclear industry has to rely on solutions developed for other industries. This is necessary, to have a large enough experience database accumulated in the use of the systems. On the other hand, the nuclear industry obviously has some very special requirements with regard to the validation of selected solutions. In the development of new systems, these requirements may be reflected to some extent if arguments are presented in a convincing way and at the right moment. One problem, in which the nuclear industry has to invest a considerable amount of thinking, concerns the adaptation of the rapid information technology development to the very long NPP lifetime. The nuclear industry would need a number of new plant projects to accomplish full utilisation the opportunities inherent in the new I&C systems.

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STRATEGIES FOR THE PREVENTION AND MITIGATION OF SEVERE ACCIDENTS

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Abstract

The currently operating nuclear power plants have, in general, achieved a high level of safety, as a result of design philosophies that have emphasized concepts such as defense-in-depth. This type of an approach has resulted in plants that have robust designs and strong containments. These designs were later found to have capabilities to protect the public from severe accidents (accidents more severe than traditional design basis in which substantial damage is done to the reactor core). In spite of this high level of safety, it has also been recognized that future plants need to be designed to achieve an enhanced level of safety, in particular with respect to severe accidents. This has led both regulatory authorities and utilities to develop guidance and/or requirements to guide plant designers in achieving improved severe accident performance through prevention and mitigation. The considerable research programs initiated after the TMI-2 accident have provided a large body of technical data, analytical methods, and the expertise necessary to provide for an understanding of a range of severe accident phenomena. This understanding of the ways severe accidents can progress and challenge containments, combined with the wide use of probabilistic safety assessments, have provided designers of evolutionary water cooled reactors opportunities to develop designs that minimize the challenges to the plant and to the public from severe accidents, including the development of accident management strategies intended to further reduce the risk of severe accidents. This paper describes some of the recent progress made in the understanding of severe accidents and related safety assessment methodology and how this knowledge has supported the incorporation of features into representative evolutionary designs that will prevent or mitigate many of the severe accident challenges present in current plants.

1. INTRODUCTION

From the early days of nuclear power plant development, the possibility of a reactor accident has been recognized. Accordingly, nuclear power plants traditionally have been designed following a concept of defense-in-depth, utilizing multiple barriers to both prevent and/or mitigate the consequences of a reactor accident. This approach typically involves the specification, by national safety authorities, of "design-basis" accidents (accidents that are postulated to occur), which the plants have to be designed to withstand. Nuclear plant designers then engineered plant systems, often assuming a single failure in the system, to prevent and/or mitigate these accidents to minimize the fission product releases to the environment. This approach has, in general, served the nuclear industry well to ensure a low level of risk to the public. Although accidents beyond the design basis accidents, which could result in large releases of fission products, were recognized as conceivable, they were considered highly improbable. Reliance was placed upon the defense-in-depth concept to minimize the likelihood and consequences of such accidents.

With the completion of early probabilistic safety assessment (PSA) studies, such as WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," in 1975, there became a growing awareness that severe accident prevention and mitigation merited further consideration. WASH-1400 also represented a significant advance in the use of PSA methods in reactor safety to identify the more likely accident sequences, and also potential vulnerabilities in plant design that could lead to failures or bypass of the multiple safety barriers. The accident at the Three Mile Island Nuclear Plant, Unit 2 (TMI-2) in 1979, served to crystallize the recognition that severe accidents needed further attention. Initially, this involved an evaluation of the capability of existing plants to tolerate a severe accident. For a number of nuclear power plants, it was found that the design-basis approach resulted in significant safety margins for the analyzed events and that these margins permitted operating plants to accommodate a large spectrum of severe accidents. However, most national authorities responsible for the safety of nuclear installations also undertook a revision of their positions with respect to "beyond design accidents". Regulatory bodies, in general, expected that evolutionary plants would achieve a higher standard of severe accident safety performance than previous designs, and have generated regulatory guidance or requirements to guide the design of future plants. For example, in the United States, the USNRC issued a policy statement on the resolution of severe accident issues in which these expectations for new designs were initially articulated. This policy statement was later followed by more detailed review standards for severe accident issues that went beyond existing requirements. At the same time, nuclear utilities recognized the need to establish a higher standard for future plants to gain acceptance of both regulatory bodies and the public. As a result, additional guidance was developed to be used by designers to address severe accident issues.[1] In Germany and France, the advisory committees Gesellschaft für Anlagen- und Reacktorsicherheit (GRS) and Institut de Protection et de Sûreté Nucléaire (IPSN) developed in a coordinated way a joint safety approach for future pressurized water reactors, which was proposed to and adopted by the national safety authorities Bundesministerium für Naturschutz, Umwelt und Reaktorsicherheit (BMU) in Germany and Direction de la Sûreté des Installations Nucléaires (DSIN) in France in 1993.[2]

In Canada, the regulatory requirements set by the Atomic Energy Control Board (AECB) had always included certain severe accidents within the design basis. The first commercial CANDU, Douglas Point, used a probabilistic approach to licensing, which *ipso facto* included some severe accidents. By the time the Pickering-A plant was licenced in the early 1970s, the AECB requirements for safety design were more deterministic, but still included addressing "dual" failures such as loss-of-coolant plus loss of emergency core cooling (LOCA+LOECC). [3] This concept was further generalized in 1980 during the licensing of the Darlington Nuclear Generating Station near Toronto: a large number of accidents combined with failures of the mitigating systems, had to be analyzed, including some severe accidents. [4] The reason such combined failures could meet design-basis accident dose limits was because of the moderator surrounding the fuel channels: it could accept and remove the fuel decay heat and prevent gross UO_2 melting even with no fluid in the fuel channels. Thus this class of severe accident was "contained" within the fuel channels. However further equipment failures, such as loss of moderator heat removal, which could lead to severe core damage (collapse of the fuel channels in the calandria) were not required to be analyzed at that time. The licensing approach in Canada is largely non-prescriptive (the model is "applicant proposes/regulator disposes"), so the AECB has not yet issued requirements for severe core damage accidents. However AECB stated clearly their expectations that the design of evolutionary CANDU designs would explicitly include both preventative and mitigative measures; at the same time AECL was responding to utility requirements for severe accidents by incorporating appropriate features (discussed below) in the CANDU 9 design. Similarly, in keeping with the philosophy used by these countries, a recent proposal by the International Atomic Energy Agency (IAEA) recommends that severe accidents beyond the existing design basis should be systematically considered and explicitly addressed during the design process [5].

In support of this changing regulatory environment, a large worldwide effort in the area of severe accident research began after the TMI-2 accident to provide the nuclear community with the technical data, analytical methods, and expertise necessary for assessing plant response to a range of severe accident scenarios, assessing the efficacy of various strategies to prevent or mitigate the consequences of severe accidents, including improved design features or accident management strategies, and assessing the consequences of severe accidents. Early containment failure, or avoidance of it, became the focal point of much of this research and is key to many of the safety requirements for new designs, while at the same time research continued to make progress in an understanding of the basic phenomenological issues associated with severe accident progression (e.g., source term, hydrogen generation, fuel-coolant interactions, debris coolability) and in improving the complex computer codes necessary to analyze severe accident issues. This research provides the foundation of information and understanding of severe accident behavior to allow plant designers to minimize or eliminate the challenges to the plant from severe accidents. As a result, for evolutionary nuclear plants, designers are able to take full advantage of the insights gained from probabilistic safety assessments, operating experience, severe accident research and accidents analyzed by designing features to reduce the likelihood that severe accidents will occur and, in the unlikely occurrence of a severe accident, to mitigate the consequences of such an accident. A recent review of trends in the development of water cooled reactors has resulted in a set of severe accident challenges that are commonly being considered in new plant designs[6]. Among these are challenges from high pressure melt ejection and direct containment heating, hydrogen production and combustion, steam explosions, both in-vessel and in the containment, and core-concrete interactions.

2. DISCUSSION

The large body of research into severe accidents initiated world-wide after the TMI-2 accident has provided not only the basis to address severe accident issues for operating plants but has also provided the knowledge and analytical tools to improve the designs of evolutionary plants to allow them to achieve a higher level of severe accident performance. The progress in understanding the challenges to containment from severe accidents and design approaches being considered to prevent or mitigate these challenges in selected evolutionary designs are presented in the following areas: high pressure melt ejection and direct containment heating; fuel-coolant interaction; debris coolability; and hydrogen combustion. In addition, progress in understanding the potential for in-vessel retention is also discussed as it provides the potential to eliminate several of the challenges to the containment from severe accidents. (The discussion in this paper will focus on the strategies utilized in the evolutionary plants in the US (CE System 80+, ABWR, AP-600), Canada (CANDU 9) and the European Pressurized Water Reactor). In many of these areas, the advances in the understanding of some of the more important severe accident challenges are often leading to common approaches being employed by designers to deal with these challenges.

2.1 High Pressure Melt Ejection (HPME)/Direct Containment Heating (DCH)

In certain reactor accidents, degradation of the reactor core can take place while the reactor coolant system remains pressurized. A degraded core left uncooled will slump and relocate to the bottom of the reactor vessel. If the reactor vessel fails, the core melt can be ejected into the containment cavity under pressure. If the material subsequently should be ejected from the reactor cavity into the surrounding containment volumes in the form of fine particles, thermal energy can be quickly transferred to the containment atmosphere, pressurizing it. The metallic components of the ejected core debris could further oxidize in air or in steam and can generate hydrogen and chemical energy that would further pressurize the containment. This process is called direct containment heating (DCH).

A number of Probabilistic Safety Assessment (PSA) studies for existing reactors, such as NUREG-1150 in the U.S. have identified DCH as one of the major threats for early containment failure for pressurized water reactors. As a result of the concern of early containment failure arising from DCH, this issue has been discussed by the international nuclear safety community for a number of years and a significant experimental and analytical program had been undertaken to support resolution of the DCH issue for existing plants. A recent review of the this large body of work is contained in the recent State of the Art Report issued by CSNI in December 1996.[7] In particular, a large program was undertaken in the U.S. which consisted of (1) integral testing at different scales, (2) separate effects testing, and (3) analytical model development and validation. This research has provided the necessary information for a defense-in-depth approach for prevention or mitigation of DCH as a threat to early containment failure for evolutionary designs.

In select evolutionary designs currently being pursued, HPME has been considered and a combination of design features contribute to the prevention and/or mitigation of the consequences of high-pressure melt ejection. A common provision in minimizing the likelihood of HPME is to ensure that any significant core damage events will occur at low pressure. Other design features contributing to the mitigation of the consequences of HPME (the effectiveness of which has been confirmed by research results) are designs of the reactor cavity to limit debris dispersal and thereby prevent interaction of core debris with the containment atmosphere, and design of robust containments to withstand the predicted loads from DCH. For reactor designs, such as the Combustion Engineering (CE) System 80+, European Pressurized Water Reactor (EPR), Advanced Boiling Water Reactor and Westinghouse AP-600, a key design feature is to incorporate reliable systems for reactor system depressurization to minimize the likelihood of HPME and to reduce the threat of DCH. A second line of defense for the CE System 80+, ABWR, AP600 and EPR designs is to incorporate reactor cavity designs that limits the fraction of core debris dispersed to the upper containment atmosphere, thereby limiting the resultant loads to the containment from HPME and containment designs that will withstand the remaining loads from DCH.

CANDU reactors include forced depressurization through opening main steam relief valves on the secondary side, as well as incorporation of diverse and independent cooling systems to provide for a similarly low probability of a severe core damage accident occurring at high pressure. Further, the design of the CANDU is such that if a total loss of heat sink at high pressure occurred, a limited number of pressure tubes would eventually fail due to over-pressure and over-heating. These failures would depressurize the reactor prior to significant fuel melting, so that any further core degradation would occur at low pressures. The discharge from these failures would be to the liquid moderator, not directly to containment in any case. Thus DCH is not an issue.

2.2 Fuel-Coolant Interactions

Fuel-coolant interaction (FCI) is a process by which molten fuel transfers energy to the surrounding coolant, leading to breakup and quenching of melt with possible formation of a coolable debris bed or, alternately, the fuel-coolant interaction could take the form of an energetic steam explosions that could challenge reactor vessel and containment integrity.

Since the quantification of the containment failure mode induced by in-vessel steam explosion-generated missiles in the Reactor Safety Study, WASH-1400 (identified in the study as the alpha-mode failure), significant progress has been made in understanding the processes and parameters that effectively limit the potential of missile-induced failure by an in-vessel steam explosion. Most recently, in June 1995, a second Steam Explosion Review Group (SERG-2) workshop was convened by USNRC to review the current understanding of the complete spectrum of FCI issues by a panel of international experts. The first Steam Explosion Review Group (SERG-1) workshop took place in 1985. The SERG-2 experts generally concluded that the alpha-mode failure issue was resolved or "essentially" resolved meaning that this mode of failure is of very low probability and of little or no significance to the overall risk in a nuclear power plant. NUREG-1524 [8] was issued in August 1996 summarizing the deliberations of the experts.

The SERG-2 experts noted that with the essential resolution of the alpha-mode failure issue, the emphasis of FCI research shifted to other issues such as mild quenching of core melt during non-explosive FCI, and shock loading of ex-vessel structures arising from explosive localized FCI. These issues are relevant with regard to assessing certain accident management strategies for operating reactors and the adequacy of certain passive system design features of evolutionary water reactors.

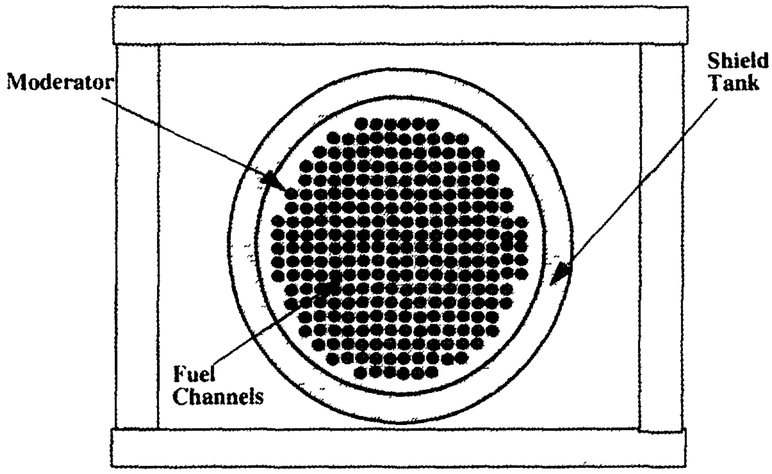

FIG. 1 Emergency heat removal in the CANDU 9

There continues a large body of experimental and analytical work ongoing on in-vessel and ex-vessel steam explosions [9], in programs such as the FARO/KROTOS program at the Safety Technology Institute of the Joint Research Center at Ispra, Italy. Although considerable progress has been made in the basic understanding of FCI, energetic steam explosions in reactor geometries and with reactor prototypic materials can not be excluded at present. Therefore, the energetics and consequential damage from localized FCI must continue to be considered and assessed in the design of evolutionary reactors. In the FARO facility, large masses (typically, up to 250 kg) of prototypic reactor melt are generated and poured into a water pool of varying depths at a range of system pressures. The FARO/KROTOS experiments tests that have been carried out show generally consistent melt quenching with either no steam explosion or very mild energetics. A complementary experimental program recently completed at Argonne National Laboratory has provided information on the energetics, including chemical augmentation, of Zr-water and Zr-ZrO_2 interactions.

Because of the uncertainties in the current understanding of fuel coolant interactions and the likelihood of steam explosions, there is no common approach among the various reactor designs to prevent this challenge. Furthermore, competing strategies for the mitigation of core-concrete interactions and retention of core debris in-vessel by ex-vessel flooding have resulted in different concerns regarding the likelihood of steam explosions. For the ABWR and EPR reactors, the designers have developed designs that attempt to minimize the likelihood of water in the reactor cavity prior to RPV failure and release of molten core debris, thereby limiting the likelihood of energetic fuel-coolant interactions. Balancing the competing interests differently, the designers of the CE System 80+ and the AP-600 do not attempt to minimize the water in the reactor cavities. In fact to attempt to minimize the core concrete interactions for the CE System 80+ and to promote in-vessel retention for the AP-600, both designers take steps to introduce water into the reactor cavities. For these designs, the reactor cavity and containment designs have been designed to be sufficiently robust to survive the calculated loading on the cavity or failure of the cavity walls. Again, the inherent features in the design of the CANDU result in less likelihood of significant melt formation. In a LOCA + LOECC in which the moderator heat removal also fails, the moderator will gradually (hours) boil away. As it does so, upper fuel channels will become uncovered, overheat, and collapse onto lower ones, potentially with some fuel melting inside the channel. The end result is a buildup of coarse debris at the bottom of the calandria, instead of formation of large suspended quantities of molten fuel. Experimental validation of this predicted behavior is required and is now underway.[10] The calandria shell is surrounded by a water-filled shield tank, which will preserve the calandria shell as a fuel debris container [11] until, after about a day, it too has boiled away (this is possible because of the thin (about 1 inch) calandria wall, the large calandria surface area and the low power density of the debris). AECL's approach in CANDU 9 has therefore been to preserve the calandria shell boundary in severe core damage accidents by adding gravity-driven makeup water, sufficient to remove decay heat, to the calandria and to the shield tank (Figure 1).

2.3 Core Concrete Interaction/Melt Debris Coolability

The eventual contact of molten core debris with the concrete in the lower containment will lead to core-concrete interaction (CCI), which can challenge the containment by (1) pressurization resulting from production of steam and non-condensible gases, and (2) base-mat melt-through. CCI is affected by the availability of water in the lower containment and the composition and amount of concrete in the basemat. Melt coolability is essential to prevent both the basemat meltthrough and the continued containment pressurization due to core concrete interaction. The common strategy being pursued by most designs is to provide a large spreading area for the core debris and means to provide water to cool the core. Unfortunately, at this time, the ability to cool the core ex-vessel has not been demonstrated. The currently active experimental research on debris coolability, called the Melt Attack and Coolability Experiments (MACE) program [12], was developed as an extension of the Advanced Containment Experiments (ACE) program under the sponsorship of a number of countries. The MACE program is intended to determine the ability of water to cool prototypic ex-vessel core debris of urania-zirconia composition. To date, this program has not be able to conclude that ex-vessel core debris can be cooled by an overlying pool of water.

The ABWR has incorporated several design features to mitigate the effects of CCI, including basaltic concrete and a drywell flooding system once debris enters the drywell. In addition, the ABWR design provides a large lower drywell floor area to provide for spreading of a molten core, thereby satisfying the EPRI design criterion of $0.02m^2/MWt$, a thick reactor pedestal wall and a lower drywell flooding system. Likewise, the CE System 80+ and AP600 have incorporated many features to mitigate CCI. These include large reactor cavity floor areas for spreading core debris sufficient to meet EPRI design criterion of $0.02m^2/MWt$ and cavity flooding systems which are designed to ensure water is available in the cavity prior to core debris entering the cavity. Similarly, for the CANDU design, if core debris penetrates the shield tank, a large spreading area is provided (approx. $O.04m^2/MWt$), the location of the reactor just above the floor of the containment (and the fact that there is no basement) will provide for a water layer on any core debris. In addition concrete composition below the reactor is chosen so as to minimize production of hydrogen.

For the EPR design, considerable design and research efforts have been directed at providing an engineered core catcher design that relies on directing the molten debris to a dedicated spreading area that is covered with both sacrificial and protective layers and incorporates cooling from both above and below to prevent stabilized melt, thereby preventing base-mat melt-through. A number of experimental programs have supported this innovative approach, including spreading experiments at the COMAS facility utilizing prototypic material[13].

2.4 Hydrogen Combustion

The major concerns regarding hydrogen are that the static or dynamic pressure loads for hydrogen combustion and detonation may pose a challenge to containment integrity or to the survival or functioning of essential safety equipment. When hydrogen combustion alone is insufficient to threaten containment integrity, combustion may still represent a significant contribution to containment loadings when considered conjunctively with other phenomena.

Research conducted world-wide over the past 17 years has extensively investigated a number of issues related to hydrogen combustion and transport during severe reactor accidents. Much of the work, performed to experimentally investigate the design and evaluation basis for reactor containment performance, focused on global deflagrations of premixed volumes of hydrogen, air and steam. Significant information exists on hydrogen combustion to assess the possible threat to containment and safety-related equipment and to allow implementation of hydrogen mitigation techniques. However, there are still issues related to a better understanding of the hydrogen combustion, particularly as it affects prediction of localized combustion phenomena such as deflagration-to-detonation transition, and the ability of current analytical tools to predict hydrogen distribution in complex geometries. Further, combustion behavior in sub-compartments is uncertain and depends on geometry and hydrogen concentration. Uncertainties remain, but the uncertainties are not considered to be of a nature to prevent implementation of hydrogen mitigation measures in designs of evolutionary water reactors[14].

Again alternate approaches are utilized to limit the challenges from hydrogen combustion. For example, the ABWR utilizes a nitrogen-inerted atmosphere within the containment to prevent hydrogen combustion, while the remaining designs incorporate large containment volumes to limit hydrogen concentrations, use layouts that minimize or avoid internal areas where hydrogen can pocket, and promote natural circulation to provide mixing of the containment atmosphere. Furthermore, for the CE System 80+, AP-600, and CANDU 9 designs ignitors are employed to limit hydrogen concentrations. In addition, in the AP-600, CANDU 9 and EPR designs, passive *autocatalytic recombiners are also employed to limit the hydrogen concentrations.*

2.5 Reactor Vessel Integrity

During the late phase of a severe accident, a significant amount of core material may relocate downward into the lower head of the reactor vessel or calandria for the CANDU. When this core material is relocated into the lower head of the reactor pressure vessel, a molten pool forms without sufficient cooling and can impose a significant heat load on the reactor vessel lower head. Post-accident examinations of the TMI-2 reactor core and lower plenum found that approximately 19,000 kg of molten material had relocated onto the lower head of the reactor vessel. Both the AP-600 and CANDU designs incorporate features that can provide cooling to the exterior of the reactor vessel and potentially maintain the core material inside the vessel. Knowledge of in-vessel and ex-vessel heat transfer phenomena to the lower head is needed to assess the ability of the reactor pressure vessel to maintain its integrity during a severe accident. A major research project in this area, utilizing prototypic material, that is providing data to assess the likely success of this approach is the OECD RASPLAV project [15] on melt pool natural convection, crust formation and growth, and heat flux distribution on the RPV lower head.

The overall objective of the RASPLAV program is to provide analytical and experimental information that can be used to assess whether, and under what conditions, molten core materials can be cooled/retained inside a reactor pressure vessel. The experimental program includes several integral experiments utilizing ceramic UO_2/ZrO_2 and metallic Zr melt of varying compositions in a slice geometry representing the lower head of the RPV, and a number of separate effects experiments. Three RASPLAV integral experiments have been performed to date, along with a number of smaller scale experiments which are providing necessary data to reach a conclusion on this concept.

3. CONCLUSIONS

Because of the extensive severe accident research efforts over the past 17 years, there is now a large body of technical data and well developed analytical tools to allow for an understanding of severe accident phenomena and to address the severe accident issues on evolutionary reactors to minimize the risk to the public. A number of the major severe accident issues can now be effectively addressed in these designs, while in other areas, the challenges from severe accidents can be minimized. As a result of this effort, evolutionary reactor designs should be able to demonstrate that they have indeed resulted in an enhanced level of safety over existing plants.

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KEY DEVELOPMENTS OF EVOLUTIONARY DESIGNS

(Session IV a)

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THE EPR — A SAFE AND COMPETITIVE SOLUTION FOR FUTURE ENERGY NEEDS

XA0053550

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Abstract

NPI, Siemens and Framatome, in co-operation with EDF and the major German Utilities, started the development of the European Pressurized Water Reactor (EPR) as an evolutionary approach. After a careful evaluation of the potential of passive safety features, this way was concluded to be superior compared to a "revolutionary" approach. The vast majority of advanced reactor designs being developed today is of the evolutionary type. The advantages to base an advanced design on the feedback of operation experience of the more than 100 nuclear power plants designed and constructed by Siemens and Framatome are outstanding. This view is shared by the German and French safety authorities which defined their preference for an evolutionary design early at the beginning of their co-operation for the definition of a common safety approach for future nuclear power plants to be built in Germany and France. In their first common set of recommendations, they gave a clear guideline regarding their point of view for requirements posed to the next generation of nuclear power plants: (1) Preference for an evolutionary design in order to derive a maximum benefit from experience; (2) Significant safety improvements by reduction of core meltdown probability and improvement of the confinement function of the containment under accident conditions; (3) Improvement of operating conditions regarding radiation protection, maintenance and human errors. Besides the French-German co-operation of vendors, utilities and authorities, the European utilities co-operate on a much broader basis for the establishment of the European Utilities' Requirements (EUR). During the development of the basic design, the EPR was continuously assessed against these EUR and it was concluded that the EPR complies with these requirements. At the end of the basic design phase at the end of 1997, all information necessary to file a preliminary safety analysis report and a reliable bill of quantities was elaborated. With this information EPR is ready to be offered on the international market. During the year 1998, a further optimisation is going on to reach outstanding low generation costs, which ensure competitiveness even against combined cycle plants.

1. INTRODUCTION

The development of new and advanced reactor types represents a big challenge to the suppliers of nuclear power plants. Both, in terms of budget and human resources, those undertakings require outmost efforts. In order to meet these challenges, the two most experienced European suppliers, Siemens and Framatome, have shared their resources by creating their joint subsidiary Nuclear Power International (NPI) and allocating this task to it. NPI started the development of a new pressurized water reactor type designed to meet the requirements of the export market in 1989.

It was obvious, that a newly designed pressurized water reactor had to feature an enhanced safety level and had to improve the economic competitiveness of electric power produced by nuclear power plants. For the engineers it became rather early obvious that these two targets could only be achieved by an evolutionary approach in order to maintain the vast experience accumulated by about 100 nuclear plants that had been built by Framatome and Siemens in the past. Maintaining references was of special importance for a product dedicated to the export market and to be realized also in countries with no or little experience in nuclear technology.

Shortly after the development had started, the first results attracted the attentiveness of the German Utilities and of Electricité de France. Both conducted their own development programs for successive units of the well-proven designs of KONVOI and N4 independently. After a peer review of the existing design approach, they decided to join their own development programs with that of NPI to the European Pressurized Water Reactor (EPR).

NPI, Siemens and Framatome, in co-operation with EDF and the major German Utilities, started the development of the European Pressurized Water Reactor (EPR) as an evolutionary approach. After a careful evaluation of the potential of passive safety features, this way was concluded to be superior compared to a "revolutionary" approach. Generally, the development of designs with radical configuration changes started at a time when passive safety features were assumed to be helpful in public acceptance discussions where slogans like "small is beautiful" and "walk-away reactor" were created. Many of these designs have disappeared from conferences and from the market in the meantime. The vast majority of advanced reactor designs being developed today is of the evolutionary type. The advantages to base an advanced design on the feedback of operation experience of the more than 100 nuclear power plants designed and constructed by Siemens and Framatome are outstanding, especially in terms of reliability, licensing risk, reference designs and through the limitation of contingencies also in terms of cost.

The collaboration between Germany and France in the nuclear industry was supported by two other projects being under way at the same time. First, the German and French licensing authorities extended their co-operation from the safety survey of existing plants in both countries to the definition of common requirements for future nuclear power plants to be constructed in either of the two countries. Second, a group of European Utilities was working on a set of requirements, defining the utilities approach for new nuclear power plants. This work addresses in particular the operation conditions of new plants aiming at the competitiveness of nuclear power against alternative energy sources in terms of power generation cost.

It was therefore recommended to orient the EPR development at both, the newly established licensing requirements in both countries and to satisfy the utility requirements of the group of major European Utilities. The first approach ensures to fulfil the requirements for obtaining a license in France and Germany with a common design, so that country specific adaptations are not needed and opening the chance for series construction of EPR in the two countries. The second ensures that also the interests of the utilities, the future operators of EPR, are considered, thus leading to fulfil the second development target of competitive power generation cost. Furthermore, the acceptance of the design is broadened, so that the standardization of the design is increased even when constructed in further countries world-wide.

At the beginning of 1995 NPI was awarded with the contract for the performance of the Basic Design for EPR. During the Basic Design Phase all organizations normally involved in the design, manufacturing, operation and licensing of nuclear installations were involved from the very beginning. On the level of the manufacturers, Framatome and Siemens acted together for the design of the Nuclear Island through their joint subsidiary Nuclear Power International. The engineering division of Electricité de France supported this through the performance of its allocated scope. In October 1997 the basic design work was finalized by issuing a comprehensive report containing all information required for a preliminary safety analysis report. A bill of quantities was established that allowed for detailed cost calculation. Subsequently, power generation cost analysis was performed confirming the EPR competitiveness with alternative energy sources. However, the margins were rather small in front of the continuously decreasing gas and coal prices on the international market. It was therefore decided to continue the development process by a further phase, the so-called basic design optimisation phase aiming at a further reduction of the power generation cost without impairing the safety level of the plant. The basic design optimisation phase started directly after the basic design had been completed is still going on until the end of 1998. The major results, however, are reported in the following.

2. SAFETY OBJECTIVES

In their first common set of recommendations, the French and German safety authorities gave a clear guideline regarding their point of view for requirements posed to the next generation of nuclear power plants:

- Preference for an evolutionary design in order to derive a maximum benefit from experience.
- Significant safety improvements by reduction of core meltdown probability and improvement of the confinement function of the containment under accident conditions.

- Improvement of operating conditions regarding radiation protection, maintenance and human errors.

Prior to the start of Basic Design for the EPR, these general objectives were defined as EPR development targets. Preliminary investigations proved that the further enhancement of the safety level from the already high level prevailing in France and Germany could be achieved at reasonable costs. The combination of several features derived from the recent development of nuclear technology, like the feedback of experience from reactors in operation, better acquaintance with core meltdown phenomena and benefits gained by using already proven technologies by an evolutionary development approach helped to fulfil this development target. Consequently, EPR is easier to operate and provides an even lesser potential source for hazards to its immediate environment than existing plants.

The major design features of EPR with regard to the two particular development targets, enhancement of safety level and competitive power generation costs can be summarised under the following headlines:

Simplification of safety systems and elimination of common mode failures by physical separation of redundant safety system trains and diverse backup functions for safety functions. This design enhances the safety level and allows in parallel for preventive maintenance measures during operation that provide cost savings in the field of operation and maintenance.

Increased grace periods for operator actions by designing components with larger water inventories and reduced sensitivity to human errors. This design enhances the safety level by smoothening transients and avoiding their propagation to accident sequences. In parallel the economic performance of the plant is improved by a reduction of the forced shutdown frequency.

Although the severe accident frequency has been further reduced by deterministic design criteria and probabilistic verification of design choices, design measures have been taken to limit the consequences of severe accidents including core melt scenarios to the plant itself. Relocation or evacuation in the plant vicinity is ruled out and the restrictions to the use of foodstuff are limited to the first year harvest.

The economic competitiveness of EPR after the basic design optimisation phase is ensured by a number of measures that address all components of the power generation cost. In particular these are:

- A preventive maintenance concept leading, among others, to an improved average plant availability of 92% during the lifetime of the plant.
- An improved efficiency of 36% due to the optimisation of the secondary side and an economiser in the downcomer of the steam generators.
- An optimised building layout to reduce building volumes, investment cost and construction time.
- A design lifetime of the plant of 60 years.
- A significant reduction of fuel cycle cost mainly by the design of high burnup cores of more than 60 MWD/kgHM with excellent neutron economy and supported by a heavy reflector.
- A large unit size that has been upgraded during the basic design optimisation phase by 15% to a thermal reactor power of 4900 MW grants favourable specific investment cost by using built-in design margins of the large core consisting of 241 fuel assemblies.

The safety objectives as above are bound in a twofold strategy. First, to improve the preventive level of the defence-in-depth safety concept by provision of redundant safety systems and diverse backup safety functions, so that the probability for the occurrence of severe accidents is further reduced from the present status.

Second, even if the probability of severe accident scenarios - up to core melt - has been further reduced, to implement additional features in order to mitigate the consequences of such accidents in a way that stringent off-site countermeasures like evacuation or relocation of the population living in the vicinity of the plant are avoided.

Consequently, the EPR design includes the following features for core melt mitigation and the prevention of large releases:

- Prevention of high pressure core melt
- Prevention of hydrogen detonation
- Prevention of the molten core-concrete interaction
- Prevention of ex-vessel steam explosion
- Maintenance of containment integrity
- The evolutionary approach and its references

An evolutionary approach keeps the references for the design choices. EPR design features are based on well proven design solutions derived from the experience of more than 1500 reactor operating years. They find their references in the operating KONVOI and N4 units.

3. REFERENCE FOR THE REACTOR COOLANT SYSTEM DESIGN

The design of the reactor coolant system is derived from the existing 4-loop pressurized water reactor plants in operation in Germany and France. The thermal reactor output of 4900 MW leads to a net rated output of in the range of 1750 MW under standard central European site conditions.

With respect to the thermo-hydraulic data of the primary circuit, the design and operating pressures and temperatures do not impose any unproven new parameters. The reactor coolant system operating pressure remains at 15,5 MPa (155 bar), the Reactor Pressure Vessel (RPV) inlet temperature of 292,5°C and the RPV outlet temperature of 330°C refer to this reference design.

The above applies also to the steam generators which are designed for a feedwater inlet temperature of 230 °C at a pressure of 7,36 MPa (73,6 bar). The heat exchange area was increased to 8171 m^2 by using smaller tubes with a diameter of 19 mm.

The fuel assemblies used for EPR are identical to those of the existing French design. Only the size of the core has been enlarged from 205 to 241 fuel assemblies for better fuel management flexibility and economy

4. REFERENCE FOR THE OVER-PRESSURE PROTECTION SYSTEM ON PRIMARY SIDE

The KONVOI solution for the primary side over-pressure protection provides one dedicated relief valve and two safety valves. The relief valve features a variable set-point as provision against pressurized thermal shock during start-up and shutdown and is isolatable. The safety valves are actuated by two spring-loaded pilot valves and discharge via the relief tank into the containment atmosphere.

The N4 design provides three identical valve stations consisting of two safety valves each, actuated by spring-loaded pilot valves. They relieve also via the relief tank into the containment atmosphere.

For EPR these designs were merged in a way to apply the three identical valve stations of the N4 design but furnishing them with the sliding set point feature of KONVOI as countermeasure against pressurized thermal shock. As additional feature, they dump via a relief tank into the IRWST and not anymore into the containment atmosphere

5. REFERENCES FOR THE OVER-PRESSURE PROTECTION ON SECONDARY SIDE

The design of the valve station on secondary side follows in principle the KONVOI arrangement. For each steam generator, one isolatable relief valve and one safety valve are provided. For EPR a second safety valve was added to the KONVOI solution mainly for redundancy reasons. The set-points are selected in such a way that safety valve response is limited to rare cases such as spurious isolation of main steam lines or hypothetical cases such as ATWS. The relief valves will be qualified for the discharge not only of steam but also of sub-cooled water.

6. REFERENCE FOR THE SAFETY SYSTEM CONFIGURATION

The selection of the safety system configuration represents the most obvious example of the benefits of the French German co-operation and the synergy provided thereby. When the engineers had reviewed the existing KONVOI and N4 solutions with respect to the newly defined safety objectives, the EPR system configuration was derived from the well proven designs in order to maintain references and to benefit from operating experience. Where new solutions are introduced, relevant verification tests will prove the suitability of the design. This way, a superposition of features from both designs could be avoided; instead the best choices of both design philosophies were selected and combined in the light of a well-defined and well-balanced catalogue of safety requirements.

The objective to further reduce the severe accident probability required a new look to the principles of safety system design. For earlier plants in Germany, the focus was laid on the redundancy of systems by deterministic considerations rather than diverse backup functions as the French case was. This latter approach shows significant advantages when probabilistic considerations gain importance.

As those probabilistic considerations were defined as one of the major EPR development targets, the combination of both design philosophies promised the requested further reduction of the core melt probability. The degree of redundancy, in addition, made preventive maintenance of the safety systems during operation much easier, thus increasing the availability and helping to reach the other design target of competitive power generation costs.

The EPR system configuration provides therefore a fourfold redundancy as known from the deterministic German design approach. The medium head injection system injects into the cold leg and is reduced to an injection pressure of 8,0 MPa (80 bar) in order to avoid overfeeding of the steam generators in case of steam generator tube leaks or ruptures. The accumulators inject on the cold leg side were they are most effective. Their pressure will be set to 4,5 MPa (45 bar). For the low head injection system the German practice of a combined injection has been selected, but at a pressure of 2,5 MPa (25) bars.

A further example in this respect concerns the residual heat removal system. KONVOI provides a four-fold redundant Residual Heat Removal system (RHRS), which in addition to its operational function of removing the residual heat during shutdown states also serves the safety grade function of the low head injection system with the recirculation and cooling function for long-term LOCA mitigation. Due to its safety related function it is located outside containment in order to allow for repair in case of demand. N4 on the other hand features separate systems for low head injection and residual heat removal. The low head injection system consists of two trains and the residual heat removal system provides two pumps for availability reasons. In order to reduce the risk of LOCA outside the containment, the RHRS is located inside.

For EPR a system configuration similar to that of KONVOI has been selected mainly by reasons to ensure maintenance during residual heat removal operation and accident mitigation where low head injection is required. In normal plant cooldown, the RHRS is used only at temperatures of the reactor coolant system below 120°C. In case of accident mitigation any two of the RHRS trains are capable of ensuring decay heat removal.

Beyond these deterministic redundancy requirements back up functions for the case of a complete loss of one of the redundant safety systems are provided. These back up functions have their benefits especially when probabilistic risk assessments for residual risk reduction are performed.

Under this aspect the combined function of the secondary side heat removal, the accumulator injection and the low head safety injection and residual heat removal system are designed to be able to replace the function of the medium head safety injection system. In a similar way, the complete loss of the low-head safety injection and residual heat removal system is backed up by the design of the medium-head safety injection system as far as the injection function is concerned. For small break LOCA the heat removal function is taken over by secondary side heat removal.

Finally, the loss of secondary side heat removal is backed up by primary side feed and bleed through an adequately designed and qualified primary side overpressure protection system.

7. REFERENCE FOR THE BREAK PRECLUSION CONCEPT

The application of the break preclusion concept has a long history in the German licensing procedure, since it was already applied to the main coolant and even to the main steam lines inside containment of the KONVOI units. The advantage of this concept is well understood since it reduces the number of whip restraints compared to the French practice. The existing German operating practice and the consideration of the international trend, supported by the development of fracture mechanics which allows a reliable evaluation of the materials and combined with advanced QA and in-service inspection methods, made the consensus possible to apply the break preclusion concept for the main coolant lines of EPR. However, the break preclusion concept application implies a necessity of a reliable primary leak detection system. The safety injection systems and the containment will still be designed to cover a potential large break of the main coolant line.

8. REFERENCE FOR THE FEEDWATER SUPPLY SYSTEMS

Also on secondary side the principle to combine a separation of functions with the provision of backup functions is applied. Different to N4, the emergency feedwater system has now only a safety function and is not further used for the operational function of start-up and shutdown of the plant. Its four trains are completely separated without active headers. Different to N4 and KONVOI, the four emergency feedwater pumps are driven by electric motors but fed by diverse sources such as the offsite power supply system, 4 large diesels, which feed, into the four train emergency power. An additional diversity is achieved by providing two small additional diesels with lower voltage level (690 V instead of 10 kV) to ensure feedwater supply even in case of total loss of the four main diesel generators.

For operational purposes, as known from KONVOI, a dedicated start-up and shutdown system is provided. Its pump is also driven by an electric motor and is for the time being not considered for any safety related operating procedures.

9. CONCLUSION

For the development of EPR, NPI is working on a basis predefined by boundary conditions set by the safety authorities through their commonly established recommendations and by the European utilities through their European Utilities Requirements (EUR).

During the basic design phase that was completed at the end of 1997, the two major targets of licensability and of competitive power generation costs were reached. Assessments show that EPR will be competitive with alternative power sources. After the basic design, the optimization of the hardware design aiming at a further reduction of investment cost and an even improved economic competitiveness was addressed. Through the basic design optimization phase and in particular due to the high rated output of the plant the specific investment cost were further reduced ensuring the competitiveness of nuclear power and providing a comfortable margin over competitive energy sources.

EPR is developed by following an evolutionary approach but providing many advanced design solutions. Through the combination of outstanding features from both, German and French origin, these new design features provide still references in German and French nuclear power plants that are successfully in operation for a long period of time. Where these references cannot be quoted as for example regarding severe accident mitigation features, extensive research and development efforts were made to prove the suitability of the design choices.

EPR features a well balanced design at an outstanding safety level and still providing competitive power generation costs, thus it will be a challenge for the competition on the international market.

XA0053551

EVOLUTIONARY DESIGN AND CONSTRUCTION — THE SYSTEM 80+™ SOLUTION TO THE COST-BENEFIT DILEMMA

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Abstract

ABB Combustion Engineering Nuclear Power provides advanced nuclear power plant designs using an evolutionary design and implementation process. Since the emergence of severe accident concerns following the Three-Mile-Island accident, the industry has faced the problem of providing improved designs that are more economical to construct and operate relative to the current generation of operating nuclear power plants. It is easy to improve plant safety in the design process, but it is not so easy to develop features that are economical while at the same time improving safety. Moreover, it is critical to add only those design features that can be implemented in actual construction programs with high confidence that they will perform as expected. The key to ABB's development and implementation of advanced reactor designs is an evolutionary process that relies on use of proven design concepts and proven components. Furthermore, design improvements are implemented gradually in actual construction programs in order to maintain very high confidence that construction schedules will not be adversely affected and that plant startup and commercial operation will proceed as expected. ABB has demonstrated the success of the above approach through the System 80® design implemented at the Palo Verde Nuclear Generating Station, the gradually improved Korean Standard Nuclear Power Plant (KSNPP) designs (based on System 80) being constructed in the Republic of Korea, and the System 80+ Standard Plant Design certified by the US Nuclear Regulatory Commission (NRC) in 1997. This paper describes the improved redundancy, diversity, and simplicity of the more significant advanced design features that were included in the System 80+ design and that are now being implemented in the KSNPP and Korean Next Generation Reactor (KNGR) programs. Examples described herein include plant safety systems, control and electrical systems, and severe accident mitigation systems. Probabilistic Safety Analyses were performed to identify those design features with the most significant impact on plant safety as measured through core damage frequency, containment reliability, and off-site radiological releases. This paper presents improvements in core damage frequency for various initiating events and identifies the advanced design features, which contributed to the improvement. The radiological doses for System 80+ are presented and compared to those for plant designs typical of those analyzed when the current emergency planning regulations were being established. The results show significantly lower offsite doses to individuals at all distances from the plant and also a significant reduction in the land area for protection against radioactive foodstuff ingestion.

1. INTRODUCTION

The international nuclear power plant market has created demands of increased safety, increased reliability, cost competitiveness, and compliance with design and licensing criteria from a number of different countries. While some plant design improvements can lead to both increased safety and decreased costs, frequently increased safety leads to increased costs – providing the plant designer with the dilemma of choosing between improved safety and increased cost.

ABB believes that a significant part of the answer to this dilemma is the use of "evolutionary" design improvement and implementation (construction) processes. In this context, an evolutionary design process means that design improvements are made in relatively small steps using proven components in order to maintain high confidence in the design itself, in the ability to quantify safety benefits and readily license the design change, and in the performance of components themselves when actually installed. An evolutionary construction process is one in which design improvements are implemented gradually to ensure that the improved systems and components perform as expected and that the construction schedule is not affected. The remainder of this paper will describe (1) examples of the evolutionary design improvements in the System 80+ Standard Plant Design [Ref. 1] and the evolutionary implementation of those changes in the ongoing KSNPP construction program,

(2) examples of the System 80+ advanced design features being included in the KNGR program, and (3) the impact of design improvements on plant safety as determined through a Probabilistic Safety Assessment for core damage frequency and offsite radiological doses.

2. DESIGN IMPROVEMENTS

2.1 Emergency Feedwater System (EFWS)

The System 80+ EFWS consists of two divisions, each with two emergency feedwater pumps and one EFWS storage tank. In each division, one pump is motor-driven and one pump is steam turbine-driven. Each pump has the capacity for full decay heat removal. Each seismically designed storage tank has enough water volume to provide accident mitigation and plant cooldown to cold shutdown conditions. A cavitating venturi is included in each division to limit the maximum flow to a ruptured steam generator. The EFWS is started automatically by a low steam generator water level signal.

Features to Increase Redundancy:

- a 100 percent increase in the EFWS pump redundancy, achieved by increasing the number of pumps from 2 to 4, which eliminates the need for cross connection piping between EFWS divisions
- two dedicated EFWS tanks, located inside the seismically designed Nuclear Annex building, eliminating the reliance on an external shared condensate storage, which may be exposed to external hazards such as tornadoes and typhoons
- increased EFWS tank capacity
- two division separation of the EFWS, and four quadrant separation of the EFWS pumps and piping, for protection from hazards such as fire, flooding, and pipe whip

Features to Increase Diversity:

- addition of piping connections to permit gravity feed from a water source, the condensate storage tank, which is diverse from the EFWS tanks
- addition of a diverse alternate AC power source, a combustion turbine generator, which can supply power to either EFWS division

Features for Simplification:

- cavitating venturis, which limit flow to steam generator pipe breaks and eliminate the need for automatic flow isolation controls
- elimination of automatically controlled valves in the cross connection piping between divisions
- elimination of safety related function for the condensate system
- addition of capability to test EFWS pumps at rated flow while the plant is operating

The improved EFWS is included in the KSNPP construction program as well as the KNGR program.

2.2 Safety Depressurization System (SDS)

The SDS rapidly vents steam from the pressurizer to permit feed and bleed cooling of the reactor core after a total loss of feedwater, and to reduce RCS pressure after a severe accident to prevent a high pressure molten core ejection. The SDS is manually actuated by the plant operators, in accordance with emergency operating procedures.

The SDS consists of two redundant piping trains from the pressurizer to the spargers in the IRWST. The SDS valves are powered from diverse electrical power sources to ensure that at least one train can be opened, and both trains can be closed, even if a complete AC electrical division failure occurs. Blowdown is to the in-containment refueling water storage tank – as shown in Figure 1.

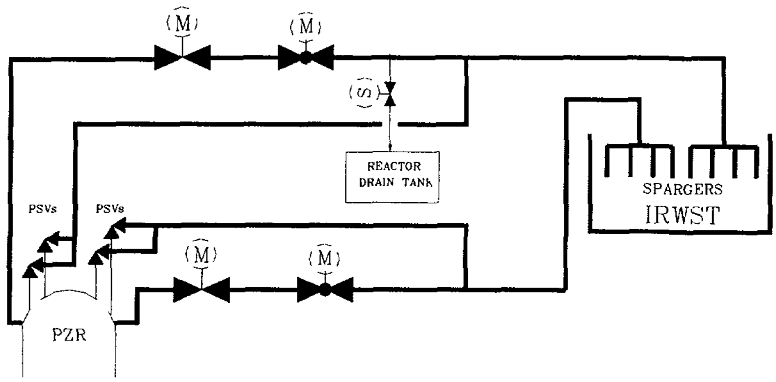

FIG. 1. System 80+ SDS

The SDS with blowdown to the containment has been implemented in the KSNPP program and, in the KNGR program, the use of power-operated safety relief valves is being evaluated.

2.3 Interfacing System LOCA (ISLOCA) Protection

The ISLOCA issue is concerned with a rupture outside containment in a system that interfaces with the RCS. Such a rupture may create a direct path for off-site radioactivity releases. The possibility of an ISLOCA for System 80+, KSNPP, and KNGR has been essentially eliminated by design features. System design pressure rating has been increased to at least 900 psig [MPa] in the SCS and in portions of other systems connected to the RCS or SCS, such as the SIS, CSS, and CVCS. The increased design pressure rating provides sufficient margin to prevent pipe ruptures, even if pressurized to the full RCS operating pressure of 2250 psia [MPa]. Other improvements include larger capacity relief valves, automatic valve closure controls, smaller vent and drain lines, and fewer intersystem connections.

2.4 Electrical Distribution System (EDS)

The EDS consists of the AC electric power system, the Class 1E AC instrumentation and control power system, the Class 1E DC power system, the emergency diesel generator system, and an alternate AC (AAC) power source. Significant advanced design features in the System 80+ EDS are:

Features to Increase Redundancy:

- two additional unit auxiliary transformers (one spare) and one additional reserve auxiliary transformer
- addition of the capability to supply power through the main generator circuit breaker to the plant AC electric power distribution system
- increased battery capacity for station blackout mitigation (two additional Class 1E batteries and two additional non-Class 1E batteries)

Features to Increase Diversity:

- addition of the AAC source, typically either a combustion turbine generator or a fifth diesel, capable of supporting one Class 1E bus of the AC electric power system plus the corresponding non-safety electrical loads
- automatic loading of the AAC source during a station blackout
- addition of turbine generator capability to remain operable and run back to house loads after a loss of load

Features for Simplification:

- elimination of the reactor coolant pump bus fast transfer from onsite to offsite power on loss of the main generator

As with the EFWS and SDS, the improvements to the electrical distribution system are included in the KSNPP construction program as well as the KNGR program.

2.5 Nuplex 80+™ Advanced Control Complex

The Nuplex 80+™ Advanced Control Complex is a computer-based, evolutionary design implemented with proven technologies. Plant-wide integration of digital I&C and man-machine interfaces is achieved while retaining the information presentation, control access, and spatial dedication formats present in existing control room designs. Prudent application of modern I&C display and control technology provides an economical, user-friendly and highly reliable advanced control complex for the System 80+ design.

The evolutionary implementation of Nuplex 80+™ advanced I&C technology is clearly demonstrated through its application to the KSNPPs being built in the Republic of Korea. Currently there are three operating KSNPPs in Korea and five in various stages of design, construction, and start-up. Though each unit has a relatively unchanged conventional control room, the I&C systems have become progressively more advanced. This has been accomplished in a conservative, evolutionary manner as illustrated in Table 1.

At the heart of the Nuplex 80+ Advanced Control Complex is the all-digital Plant Protection System (PPS) which automatically initiates reactor trip and starts safety systems in response to plant transients and accidents. Initiation signals from the PPS are sent to the reactor trip switchgear and the engineered safety features component control system to actuate protective functions.

Diversity is built into Nuplex 80+ by maintaining separation in both hardware and software between safety-grade systems and non-safety-grade systems. This diversity ensures protection against software common mode failures. A detailed safety analysis was performed to ensure that all regulatory criteria can be met even if it is assumed that there is a common mode failure of the safety-grade software. In a few instances where it was not practical to provide complete diversity between the safety and non-safety systems all the way to the actuated component, a few manual switches and hardwired circuits were added for manual reactor trip and direct operation of critical safety system components. While similar results are expected for the all-digital PPS being constructed in Korea, the detailed analysis for the System 80+ design showed that direct initiation was required for only manual reactor trip and one train of (1) safety injection system pumps and valves, (2) EFWS pumps and valves, (3) containment spray system pumps and valves, (4) main steam isolation valves, (5) containment air purge isolation valves, and (6) CVCS letdown isolation valves.

Simplification of the information presented to the operator and reduced cabling costs for Nuplex 80+ are summarized in Figure 2. Similar benefits and lower costs are expected for the KNGR design.

Table 1 Major Advanced I&C Features for System 80

Design Feature	YGN 3&4 (1987)	UCN 3&4 (1991)	YGN 5&6 (1995)	UCN 5&6 (1996)
PLCs (Diverse Protection; Rod Drive Control)	X	X	X	X
Digital Balance of Plant Control System	X	X	X	X
Video Display Unit Monitoring	X	X	X	X
Non-Safety Field Multiplexing		X	X	X
Redundant Digital NSSS Controls			X	X
UNIX Advanced Workstations			X	X
Digital Plant Protection System				X
Digital Engineered Safety Features Actuation System				X

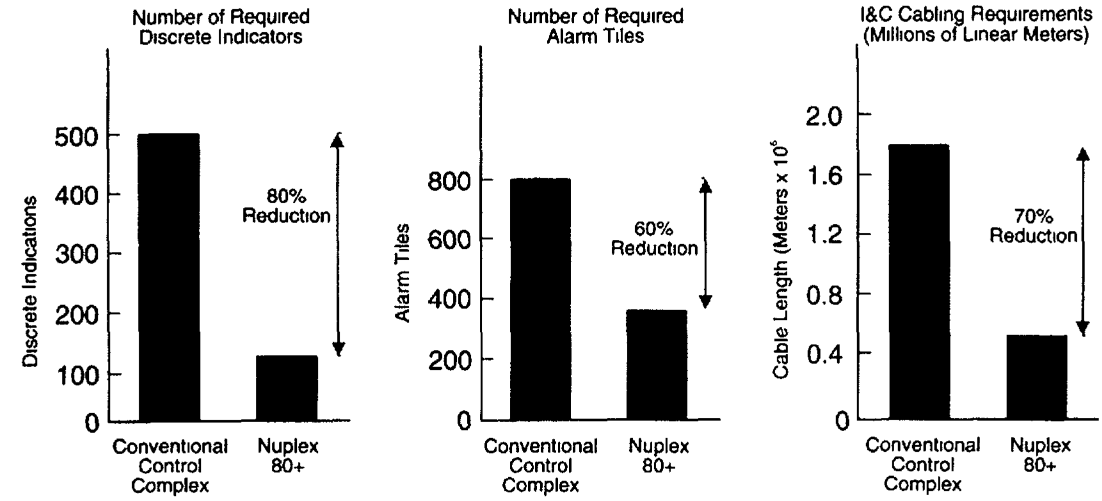

FIG. 2. Reductions in Control Room Instrumentation and Cable

2.6 Safety Injection System (SIS)

The System 80+ and KNGR SISs include four high-pressure SIS pumps that take suction from an in-containment refueling water storage tank (IRWST). The system is started automatically by either low reactor coolant system (RCS) pressure or high containment pressure. The SIS also includes four medium-pressure SIS tanks that inject water passively when RCS pressure drops below the nitrogen charging pressure in the tanks. The SIS injects borated water through four direct vessel injection (DVI) nozzles on the reactor vessel upper cylindrical shell. The SIS includes two parallel injection paths to the RCS hot legs for long-term core cooling following loss of coolant accidents (LOCAs). In addition, the KNGR program is evaluating fluidic diodes to control the rate of injection from the SITs. Significant advanced design features in the SIS are:

Features to Increase Redundancy:

- a 100 percent increase in the high pressure safety injection pump capacity, achieved by increasing the number of pumps from 2 to 4;
- DVI, reducing the loss of injection water during cold leg pipe breaks;
- four quadrant separation of the SIS pumps and piping, for protection from hazards such as fire, flooding, and pipe whip.

Features to Increase Diversity:

- addition of a diverse alternate AC power source, a combustion turbine generator, which can supply power to either SIS division.

Features for Simplification:

- continuous suction from the IRWST, eliminating the need for an automatic switch-over from an external tank to the containment sumps;
- elimination of cross connected and headered pump discharge piping;
- elimination of the need for low pressure safety injection pumps;
- elimination of automatically controlled isolation valves in the pump minimum flow recirculation lines;
- increased design pressure in portions of the system, outside containment, which could be exposed to full RCS operating pressure;
- addition of capability to test SIS pumps at full flow while the plant is operating.

2.7 Containment Spray System (CSS)

The System 80+ and KNGR CSSs consist of two divisions. Each division includes a containment spray pump that takes suction from the IRWST, a dedicated heat exchanger, piping, valves, and instrumentation. The CSS injects water through a large number of spray nozzles in several piping rings mounted on the upper containment steel shell. The system is actuated automatically by high containment pressure. Significant advanced design features in the System 80+ CSS are:

Features to Increase Redundancy:

- a 40 percent increase in the containment spray pump capacity, achieved by increasing the pump design flow
- pumps which are identical and interchangeable with the shutdown cooling pumps
- addition of dedicated containment spray heat exchangers, which adds redundancy for decay heat and containment heat removal
- addition of pump minimum flow heat exchangers
- addition of a piping connection between the CSS and the plant exterior which can be used to replenish the IRWST water supply for long term containment spray after a severe accident

Features to Increase Diversity:

- addition of a diverse alternate AC power source, a combustion turbine generator, which can supply power to either CSS division

Features for Simplification:

- continuous suction from the IRWST, eliminating the need for a switch-over from an external tank to the containment sumps
- addition of capability to test CSS pumps at rated flow while the plant is operating
- increased design pressure in portions of the system, outside containment, which could be *exposed to full RCS operating pressure*
- elimination of automatically-controlled isolation valves in the pump minimum flow recirculation lines
- elimination of reliance on the shutdown cooling heat exchangers for containment spray cooling
- elimination of spray chemical addition subsystem for post-LOCA containment water pH control

2.8 Component Cooling Water System (CCWS)

The System 80+ and KNGR CCWSs consist of two separate divisions, each with two pumps and two heat exchangers, piping, valves, and instrumentation. Each pump and heat exchanger has the full capacity to remove NSSS heat loads during normal plant operation. Each division has the full capacity to remove NSSS heat loads during plant cooldown or post-accident operation. Significant advanced design features in the System 80+ CCWS are:

Features to Increase Redundancy:

- a 100 percent increase in the component cooling water pump capacity
- a 100 percent increase in the component cooling water heat removal capacity
- two additional CCWS paths for reactor and containment heat removal, through the addition of dedicated containment spray heat exchangers
- two division separation of the CCWS in the Nuclear Island structures for protection from hazards such as fire, flooding, and pipe whip

Features to Increase Diversity:

- addition of a diverse alternate AC power source, a combustion turbine generator, which can supply power to either CCWS division

Features for Simplification:

- addition of capability to test CCWS pumps at rated flow while the plant is operating
- increased relief capacity in portions of the system, outside containment, which could be exposed to full RCS operating pressure

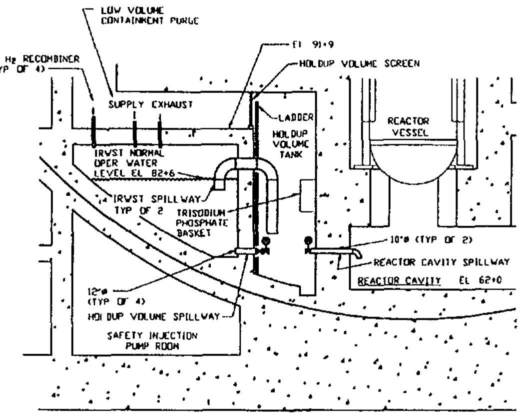

FIG. 3. System 80+ Lower Containment Region including Cavity Flooding System

2.9 Cavity Flooding System (CFS)

The CFS partially fills the reactor cavity with water during severe accidents to cool molten core material that would be released if the reactor vessel lower head fails. The system prevents the molten core debris from ablating the cavity concrete and breaching the containment pressure boundary. The CFS is manually actuated by the plant operators in accordance with accident management guidelines.

The System 80+ and KNGR CFSs consist of four spillways from the IRWST to the holdup volume tank (HVT) and two spillways from the HVT to the reactor cavity. The HVT provides an intermediate volume to prevent inadvertent flooding of the reactor cavity due to operator error. Each spillway has one motor-operated isolation valve. The CFS valves are located in the holdup volume and are designed to operated while submerged. Each of the four valves from the IRWST to the HVT is powered from different Class 1E buses. The System 80+ CFS is shown on Figure 3. As an additional enhancement, passive isolation valves are being evaluated and considered for inclusion in the KNGR design.

2.10 Hydrogen Mitigation System (HMS)

The HMS burns hydrogen in the containment atmosphere at low concentrations to prevent buildup to detonatable levels. The plant operators initiate frequent hydrogen burns throughout the containment to preserve the operability of safety systems and the integrity of the containment. For the System 80+ design, eighty (80) hydrogen igniters are provided in pairs at forty locations throughout the containment. The igniters of each pair are powered by separate Class 1E divisions of the EDS and they can also be powered by the AAC source. The igniters are placed at locations in the containment which are most likely to be close to sources of hydrogen generation or collection (general locations are shown in Figure 4). The use of passive autocatalytic recombiners (PARs) is being considered as a complement to hydrogen igniters in the KNGR design.

3. SYSTEM 80+ REDUCED CORE DAMAGE FREQUENCY AND REDUCED OFF-SITE RADIOLOGICAL IMPACTS

Design advancements incorporated into the System 80 and System 80+ plant designs, including those summarized above, have resulted in lower core damage frequencies, higher containment reliability, and correspondingly lower large release frequencies and off-site doses. The evolutionary improvement process began with ABB's 1300 MWe System 80 plant, is being continued in the System 80 KSNPP program, and is being further enhanced in the System 80+ and KNGR programs.

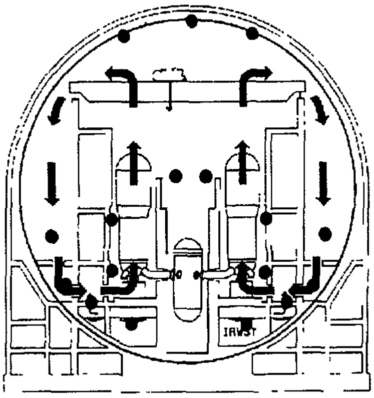

Figure 4. Hydrogen Igniter Locations

Table 2 shows analytical results. The total CDF decreased from <1.7E-4 events/year to <3.5E-6 events/year as the design evolved from the 1300 MWe System 80 design to System 80+ design. This improvement was the result of implementing the advanced design features mentioned above. The most significant design contributors to the reduction in CDF are shown in Table 3.

Severe accident mitigation features such as the use of hydrogen igniters, reactor cavity flooding, and a very conservative reactor cavity design (strong walls, thick basemat, and "tortuous" vent path to the upper containment region) resulted in a large off-site release (0.25 Sv) frequency of <5E-8 events/year for a site boundary radius of 800 meter and <6E-8 events/year for a site boundary radius of 300 meter. In addition, the mean individual Total Effective Dose Equivalent (TEDE), weighted over all core damage events, for an 800 meter site boundary is only 0.0052 Sv. The frequency-weighted Cs-137 activity release to the environment for events contributing to over 89% of the total CDF (i.e., over 97% of the early releases) is only 0.051 TBq. The maximum Cs-137 activity release for an event contributing only 0.3% to the total CDF was 17 TBq.

The Nuclear Energy Institute and the Electric Power Research Institute are also conducting a program to develop guidance for (1) assessing the effectiveness of the containment mitigation function during a severe accident in the context of emergency planning and (2) predicting the off-site doses consistent with the US Proctective Action Guidelines (PAGs) for initiation of emergency response. The PAG dose guidelines are 0.01 Sv TEDE and 0.05 Sv to the thyroid [Ref. 2]. For the System 80+ design, a severe LOCA was analyzed with the following conservative assumptions: (1) a severely damaged core and consequent reactor vessel failure, (2) only one train of containment spray operating and conservative crediting of spray hygroscopicity, (3) a maximum containment leakage rate of 0.5% volume/day, and (4) a median dose analysis using limiting US meteorological conditions. The resulting 24-hour median doses at the site boundary for System 80+ are 0.003 Sv TEDE and 0.027 Sv to the thyroid. These very low off-site doses are the result of improved severe accident mitigation systems and increased containment reliability.

Table 2 Reduced Radiological Impacts for System 80 and System 80+

Criteria	System 80 (1300 MWe)	System 80 (1050 MWe)	System 80+ (1400 MWe)
Core Damage Frequency (events/year)	Internal Events: < 8E-5 External Events <5E-5 Shutdown Risk: <4E-5 •Total CDF: <1.7E-4	Internal Events: <7.7E-6 External Events <2.5E-5 Shutdown Risk: <1.5E-6 • Total CDF: <3.4E-5	Internal Events: <1.7E-6 to 1E-7 External Events: <1E-6 Shutdown Risk: <0.8E-6 • Total CDF: <1.9E-6 to 3.5E-6
Large Off-site Release Frequency	For 0.25 Sv/24 hr: < 1.1E-5 events/year @ 900m	<8E-6 events/year @ 300m (dose not available)	For 0.25 Sv/24 hr: <5E-8 events/year @ 800m and <6E-8 events/year @ 300m

Table 3 Contributors to CDF Reduction

Internal Initiator	Major Contributors to CDF Reduction	CDF Reduction Factor
Large LOCA	• 4 high-pressure SIS pumps • IRWST • AAC source • CCWS	16
Medium LOCA	• 4 high-pressure SIS pumps • IRWST • AAC source • CCWS	12
Small LOCA	• 4 high-pressure SIS pumps • 4 EFW pumps • AAC source • CCWS	45
MSLB	• 4 EFW pumps	430
SG tube rupture	• 4 high-pressure SIS pumps • 4 EFW pumps • AAC source	37
Transients	• 4 EFW pumps • SDS (feed&bleed) • AAC source	21
Loss of Offsite Power incl. SBO	• AAC source • 4 EFW pumps • 6 vital batteries	1350
ATWS	• 4 EFW pumps • Centrifugal charging pumps • SDS (feed & bleed)	100
ISLOCA	• Higher system design pressures	9

A review of dose as a function of distance from the plant, as seen in figure 5, shows that for System 80+ the potential dose to an individual is reduced by more than two orders of magnitude at all distances from the plant relative to the WASH-1400 pressurized water reactor. The improvement relative to a typical currently operating plant (Zion) is more than one order of magnitude at all distances.

Radiation exposure to an individual can also result from ingestion of foodstuffs. In the US, the emergency response for the ingestion pathway is triggered by projected ground concentrations of several radionuclides and associated projected radiation doses. The "limiting" pathway (i.e., that used to establish the current US requirement for a 50-mile radius Ingestion-Pathway Emergency Planning Zone, or EPZ) is the dose to the infant thyroid via the drinking of contaminated milk.

Using the conservative 80-90th percentile meteorological data set from the US ALWR Program (which tends to produce high concentrations in the air and on the ground), it was found that the required distance for the ingestion-pathway EPZ would be only about 18 miles with a dry deposition velocity (DDV) of 1 cm/s and only about 12 miles with a deposition velocity of 0.3 cm/s. The former DDV value was used by the NRC in NUREG-0396 and the latter value was used by the NRC in NUREG/CR-4551 [Ref. 4]. This indicates that a substantial reduction in land area subject to detailed emergency planning for the ingestion pathway (about a factor of ten) is warranted for the System 80+ design. Note that this analysis of the ingestion pathway very conservatively addresses only the change in ingestion dose vs. distance *given that a severe accident has occurred*, and says nothing about the substantial reduction in core damage frequency discussed above.

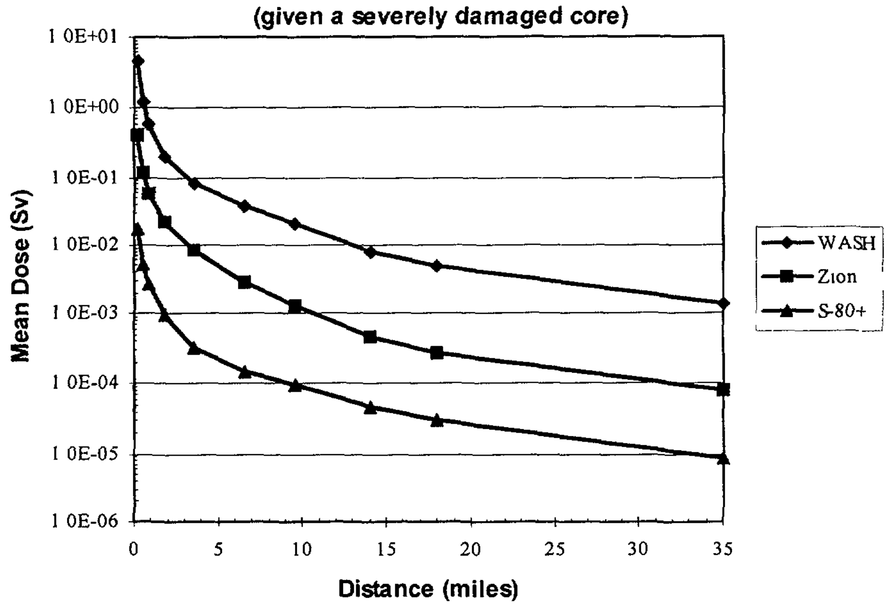

Figure 5. Mean Individual Dose vs. Distance (given a severely damaged core)

4. CONCLUSION

Development and implementation of advanced design features through an evolutionary design and implementation process is being demonstrated in the System 80+, KSNPP, and KNGR programs. This process has ensured that design features are implemented in a cost-effective manner and they perform as expected during plant startup testing and commercial operation. In addition, the advanced design features provide significant increases in safety. Reduced off-site dose predictions result from lower core damage frequency, improved safety systems, and improved containment reliability. The potential dose at any distance from the plant is reduced by more than two orders of magnitude relative to the plants analyzed when the NRC established current emergency planning requirements and long-term ingestion pathway contamination is also greatly reduced.

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DESIGN FEATURES OF APWR IN JAPAN

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Abstract

Development of the Advanced Pressurized Water Reactor (APWR) was executed in the Improvement and Standardization Program which was organized by the Ministry of International Trade and Industry, Japanese utilities (Hokkaido, Kansai, Shikoku, Kyushu Electric Power Companies and the Japan Atomic Power Company) and manufacturers (Mitsubishi Heavy Industries and Westinghouse Electric). Improvements in terms of safety, reliability, operability, maintainability and economy have been incorporated based on comprehensive evaluations of both the advanced technologies available today, and the experience associated with construction and operation of current PWR plants. The main design improvement features applied in APWR include a core design that contributes to effective use of uranium resource, safety enhancement in the engineered safeguard system, and reliability enhancement in the reactor internal structures. This paper briefly describes the main features of the APWR design focusing on the following two items: the radial reflector, which enhances reliability of the reactor internal structures as well as neutron economy in the core region; and an advanced accumulator, which enhances Emergency Core Cooling System (ECCS) reliability and contributes to system simplification due to passive low pressure injection function.

1 INTRODUCTION

In Japan, twenty-three PWRs are currently in operation. Experience in operating the early PWRs has dictated future plants might be improved, particularly in the areas of plant availability, safety, and occupational radiation exposure. So, since the mid-1970s, the Japanese government and industrial groups have worked together towards these objectives and started the Improvement and Standardization Program for Light Water Reactors. The fruits of these efforts can be seen in the APWR that is an advanced standardized plant design with higher reliability and safety characteristics. The APWR design was executed by the joint program of the PWR utilities (Hokkaido, Kansai, Shikoku, Kyushu Electric Power Companies and the Japan Atomic Power Company) and manufacturers (Mitsubishi Heavy Industries and Westinghouse Electric). The APWR is a plant design that incorporates outstanding improvement in safety, reliability, operation, maintenance and economy. The first APWR is to be adopted in Tsuruga power station unit 3 and 4 of The Japan Atomic Power Company.

2 DESIGN FEATURES

The principal specification of the APWR is shown in Table 1.1 together with that of a current 4-loop PWR plant for comparison.

Table 1.1 Principal specification of APWR

ITEM	APWR	Current 4-loop PWR
Electrical Output	approx. 1530 MWe	1180 MWe
Core Thermal Output	approx.4450MWt	3411MWt
Fuel Type	Advanced 17x17	17 x 17
Fuel Assemblies	257	193
Fuel Assembly Effective length	approx. 3,7 m	approx. 3,7 m
Core Load	approx. 120 MTU	approx. 89 MTU
Reactor Vessel inner diameter total height	approx. 5,2 m approx.13,6 m	approx. 4,4 m approx.12,9 m
Steam Generator heat transfer surface area	6500 m^2	4870 m^2
Loop Flow Rate	appr. 2,6 x 10^4 m^3/h/loop	appr. 2,0 x 10^4 m^3/h/loop
Steam Turbine	TC6F54	TC6F44
Reactor Containment	PCCV	PCCV
Engineered Safeguard System	4 trains (Mechanical)	2 trains
Emergency Water Storage	inside Containment	outside Containment

One of the important concepts of the APWR is the large power rating which decreases the construction cost per electric generation capacity. Though the electric output was planned as approximately. 1420MWe at the early stage of basic design, it was uprated to approximately. 1530MWe as a result of design progress, with efficiency improvement of the steam turbine and reactor coolant pumps, and without any change in the system configuration or main components.

The APWR core consists of 257 fuel assemblies of an advanced 17x17 type and incorporates flexibility to meet future requirements such as operation with mixed oxide fuel (MOX) in more than 1/3 of the core and high burn-up fuel of more than 55GWd/t. The inner diameter of the reactor vessel is approximately 5,2 m in order to accommodate 257 fuel assemblies.

Major components such as reactor internals and steam generators are designed to achieve high reliability taking into account actual operating experience of current PWR plants, including consideration of ageing degradation mechanisms. One of the most outstanding features of the APWR is the adoption of the radial reflector, which is made of stainless steel ring blocks and contributes to simplification of the reactor internal structures.

Enhanced safety systems appropriate for an advanced LWR coming into operation early in the 21st century are introduced. The enhanced features of these systems increase redundancy and diversity through the adoption of four mechanical subsystems in the safety injection system and containment spray system. They also require less operator action in case of abnormal events due to the emergency water storage tanks located inside the containment. In addition, the APWR adopts the original passive technology in the accumulator design called "advanced accumulator", which contributes to simplification of the ECCS design.

State of the art electronics, including digital protection and control systems, and an advanced control board are used to improve man-machine interface (MMI). Additional advanced technologies have been incorporated to facilitate operation and maintenance of the plant and to reduce occupational radiation exposure (ORE), especially during the periodic refuelling and maintenance outages.

Severe accident measures for APWR are also planned, based on accident management plans for existing PWR plants and the latest R&D information in the world.

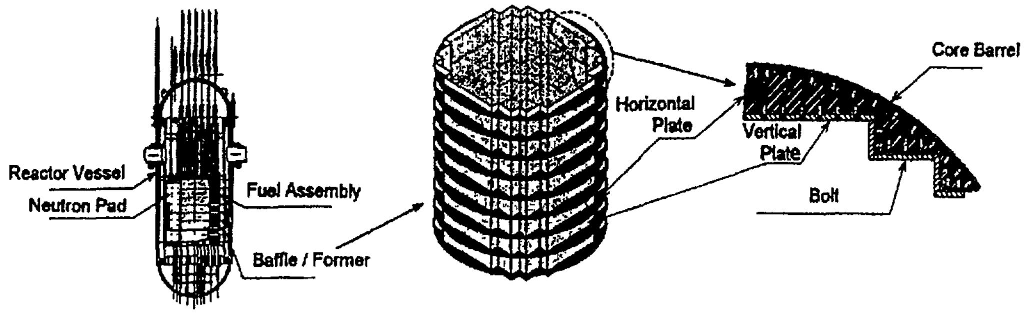


FIG. 2.1 Baffle/Former in current PWR

This paper focuses on the following two key specific features that characterize the APWR design and accommodate to Japanese PWR utilities' requirement to enhance safety and reliability and to improve economic aspects.

- Radial reflector, which enhances reliability of reactor internal structures as well as neutron economy in the core region.
- Advanced accumulator, which enhances ECCS reliability and contributes to system simplification due to passive low-pressure injection function.

2.1 Radial Reflector

One of the advanced features of the APWR is the radial reflector. In current PWR plants in Japan, the reactor internal structures have generally had an excellent operating experience; however, there remain uncertainties regarding the long-term behaviour of the materials used in the baffle because of the severe radiation environment. In case of new plants, a longer design life and a higher assumed capacity factor increase the potential of operational issues surfacing late in life.

For the APWR, therefore, the radial reflector was developed as an alternate concept of the baffle structure. The basic concept of the radial reflector was the structural simplification for reliability and improvement of neutron economy. The radial reflector design feature and supporting evaluations are described in the following.

2.1.1 Structure and Configuration

In current PWRs in Japan, the core is supported by the lower internals. The lower internals consist of the core barrel, the core baffle, the neutron shield pads, the lower core plate, support columns, and the core support plate which is welded to the core barrel. The core baffle forms a cavity that contains the fuel assemblies; it consists of vertical plates that are bolted to the horizontal plates called formers. The formers are in turn bolted to the core barrel. The baffle/former structure is shown in Figure 2.1.

The APWR radial reflector is shown in Figure 2.2. The design consists of eight stacked ring blocks. Each block is machined from a single forged 304 stainless steel ring. The interior of each

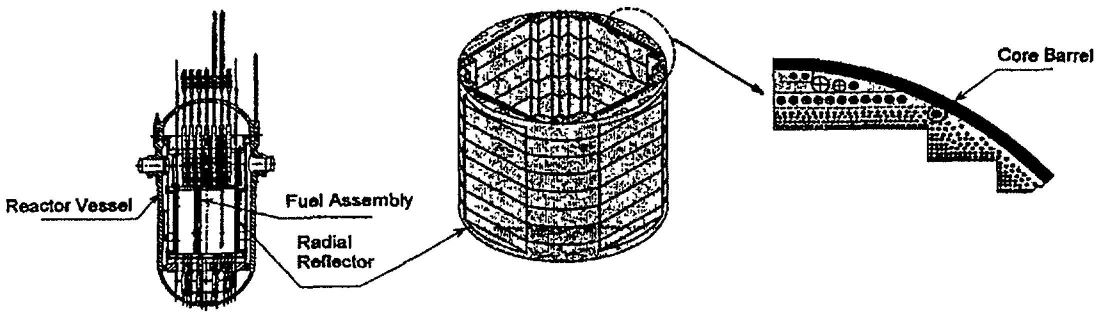


FIG. 2.2 Radial Reflector in APWR

block has the shape of the core cavity. The exterior is circular with four flat regions; the top and bottom blocks also incorporate a circular flange.

The interface gaps between the blocks are located opposite the fuel assembly grids in order to preclude any possible leakage flow impinging on the fuel rods.

Each block has about 1600 flow holes, each with a diameter of 20 mm in order to cool the blocks heated by gamma rays. Flow hole size and arrangement are decided considering the cooling effect and the neutron reflecting effect. The flow holes are orificed in the bottom block only. This arrangement results in the pressure of the core cavity being higher than in the flow holes.

There are four corner pins between each block; these are shrunk into the lower block with clearance in the upper block. The complete radial reflector is aligned to the core barrel with four horizontal pins and customized inserts at the circular flanges of both the bottom and top blocks; this is similar to the design currently used for aligning in the upper internals to the core barrel. The stacked blocks are also fastened to the lower internals with eight tie rods.

Neutron pads that are installed in the current PWRs are eliminated in the APWR because the neutron exposure rate is decreased sufficiently by the radial reflector.

2.1.2 Coolability Evaluation

The arrangement of the flow holes was optimized to obtain an even temperature distribution and to minimize peak temperature. Thermal hydraulic analysis confirmed that the peak metal temperature is lower than 330°C and remains below the target of 343 °C for prevention of coolant boiling.

2.1.3 Inspectability and Maintainability

A study was performed to assess operational aspects of the radial reflector; this included a comprehensive assessment of all known degradation mechanisms and their potential effects.

The only fasteners of the radial reflector are the eight tie rods that hold the blocks. The total number of the fasteners, which includes the few bolts used at the alignment point with the core barrel, is less than fifty. However, all threaded portions are located out of core region. And there are no welds in the design.

Based on these design features, it was assessed that no degradation mechanism that threatens the design life can be expected. Nevertheless, the radial reflector is designed to be removable from the core barrel considering inspectability and maintainability.

2.1.4 Design Verification

In order to verify the radial reflector coolability, a flow test was conducted using a 1/8 sector full scale model as shown in Figure 2.3.

The purpose of this test is to measure inlet flow rate distribution into flow holes. As the test result, it was confirmed that inlet flow distribution is very flat and satisfies design requirement.

Furthermore, the integrity against flow induced vibration was qualified by the 1/5 integrated scale model hydraulic test.

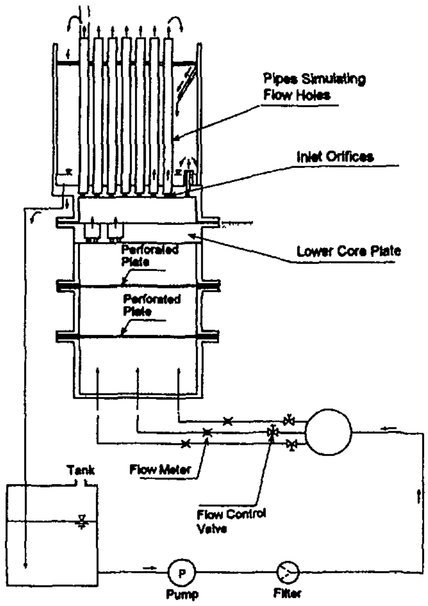


FIG. 2.3 Test Loop of Flow Distribution of Radial Reflector Flow Holes

2.1.5 Design Evaluation

An extensive design, analysis and testing was conducted for APWR radial reflector. The design evaluation is as follows:

- Threaded portions are less than fifty and are only located out of the core region. There are no welds in the design. As a result, high reliability will be expected even for a longer design life.
- Neutron irradiation on the inner surface of the reactor vessel can be reduced by approximately one-third of the current 4-loop design without neutron pads.
- The effect of the fuel cycle cost reduction is approximately 1% compared with the current 4 loop PWR

2.2 Advanced Accumulator

Improvement of the APWR engineered safeguard system includes the following three major aspects of safety enhancement:

- Mechanical 4 subsystems of engineered safeguard system
- Emergency water storage inside containment
- Advanced accumulator

These features contribute to reduction of core damage frequency through enhancement of ECCS reliability. Figure 3.1 shows the outline of the APWR engineered safeguard system.

This paper describes the third feature of the APWR engineered safeguard system; the advanced accumulator, which contributes to both safety and economy enhancement, because it can compensate the role of conventional low head injection system composed of active components such as pumps and valves with only passive components, thereby eliminating the low head injection system.

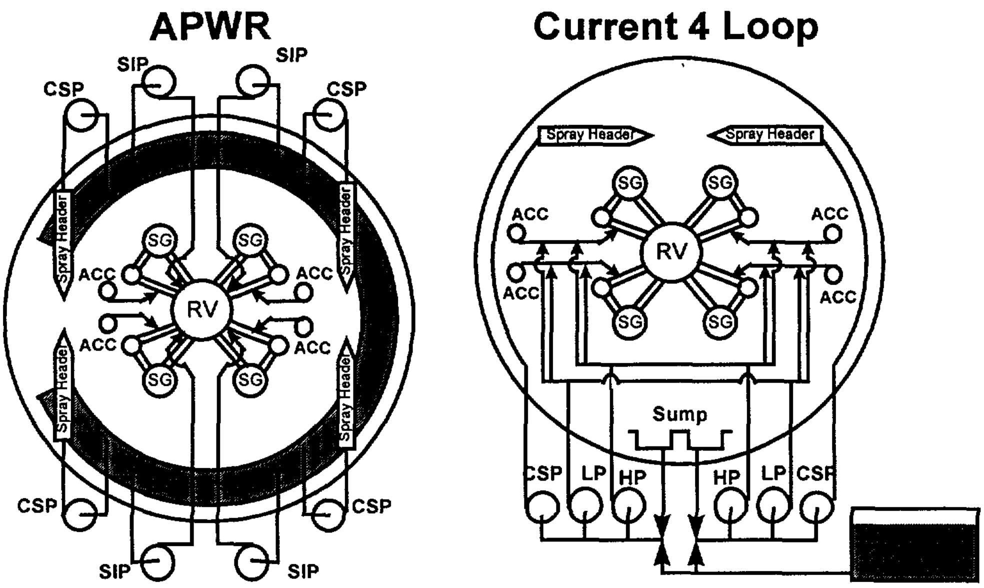
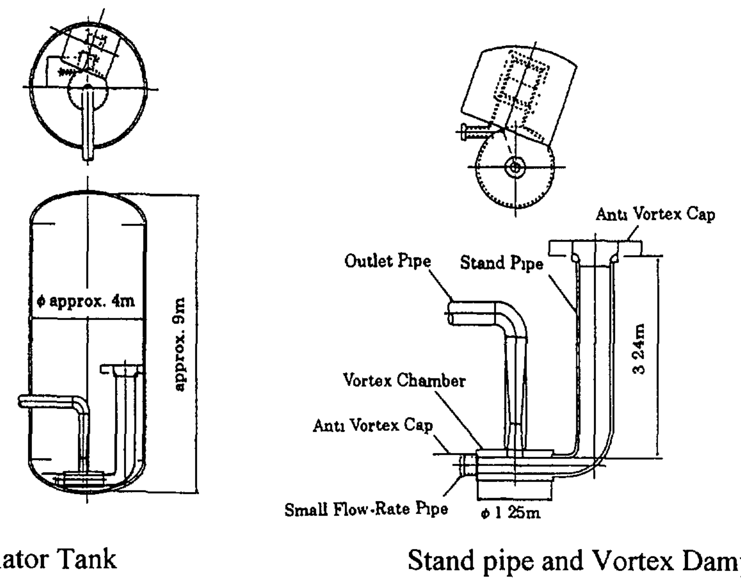


FIG. 3.1 Configuration of APWR Engineered Safeguard System

2.2.1 Outline of the advanced accumulator

In the current PWRs in Japan, four accumulator tanks are provided inside the containment. The tank contains high concentration boric acid water and is normally pressurized and maintained at approx.40kg/cm^2 by means of nitrogen gas. The accumulator tanks are isolated from the RCS by a check valve to prevent a back flow of reactor coolant. If a Loss of Coolant Accident occurs, and RCS pressure drops below that of the accumulator tank, the accumulator automatically and passively injects boric acid water into the core via the cold leg. Because the accumulator has no active part in its system, i.e. the drive force of water injection is pre-charged nitrogen gas pressure, it is categorized as a passive component with high reliability.

The advanced accumulator of the APWR injects boric acid water by pre-charged nitrogen pressure inside the tank in the same way as the conventional type accumulator. The main modification is that a stand pipe and a vortex damper are installed inside the tank to change injection flow rate utilizing "Fluidics" technology as described later. Figure 3.2 shows the outline of the advanced accumulator tank and its internal structures.



Accumulator Tank　　Stand pipe and Vortex Damper

FIG. 3.2 Outline of the advanced accumulator

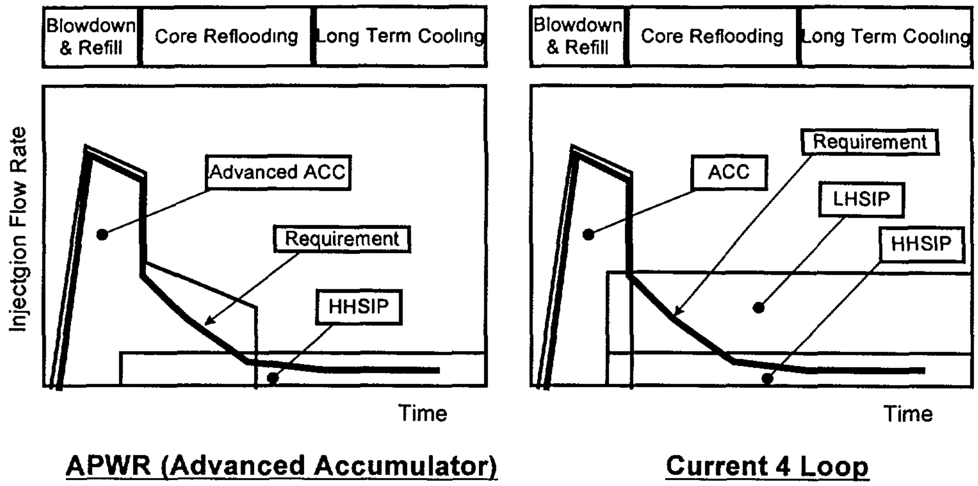


FIG. 3.3 Flow Injection Characteristics of Advanced Accumulator

2.2.2 Flow characteristics of Advanced Accumulator

While emergency core injection flow during a Loss of Coolant Accident (LOCA) is shared between high-head safety injection pumps, low-head safety injection pumps and accumulators in the current 4-loop PWR design, the APWR design allows elimination of low-head safety injection pumps as a result of introduction of the advanced accumulators with improved flow injection characteristics.

At the early stage of a large break LOCA, a large amount of cooling water must be provided quickly to start core reflooding as early as possible. But when core reflooding begins, relatively small amount of cooling water is needed to quench core and to remove decay heat. In the conventional design, accumulators supply cooling water with large flow rate at the early stage, and then relatively small flow rate injection at the later stage is performed by both low-head and high-head injection pumps.

The advanced accumulators inject the cooling water into the core at a high flow rate similar to the conventional accumulator design at the early stage of Large Break LOCA. However, the advanced accumulators are designed to then passively change injection flow rate relatively smaller at later stage. Thus fulfilling the role of the conventional low head injection pumps. After core reflooding is terminated and the accumulator tanks have emptied, long-term core cooling is performed by high head injection pumps, which has enough capacity to remove decay heat from core.

Figure 3.3 shows the flow characteristics of the advanced accumulator compared with that of conventional type.

2.2.3 Flow change Mechanism

A vortex damper and a stand pipe are installed inside the advanced accumulator to change injection flow rate without any active components. When the water level is above the top of the standpipe, water enters the vortex damper through both the inlet at the top of the standpipe and at the side of the vortex damper. Because the flow is smooth inside the vortex damper, cooling water is injected into core with large flow rate in this period. When the water level drops below the top of the standpipe, however, the water enters the vortex damper only through the side inlet, which is tangential to the damper. In this phase a vortex occurs inside the damper which increases the flow resistance, and the flow rate is reduced as a result. Thus, with this system the flow rate is changed with a simple and passive mechanism without any active components. Figure 3.4 shows the above mechanism of the advanced accumulator.

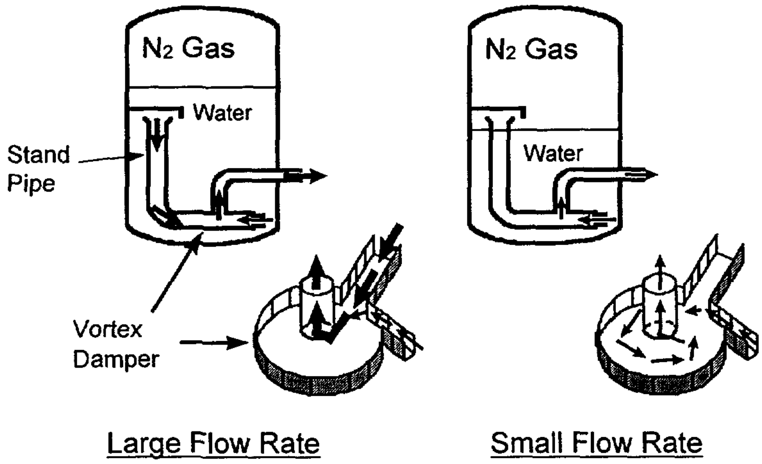


FIG. 3.4 Mechanism of Advanced Accumulator

2.2.4 Simplification of System Design

Due to adoption of the advanced accumulators, the conventional low head injection pumps can be eliminated in the APWR design. However, the low head injection pumps of the current PWR have also another function, as a residual heat removal system. So, a reorganization of system configuration was considered. Since function of containment spray and residual heat removal is not required simultaneously, and pumps of these systems have similar performance, containment spray pumps and residual heat removal pumps can be combined. Thus, a significant increase in number of safety-grade pumps has been avoided even though mechanical 4 subsystem is adopted in the APWR. A comparison between the engineered safeguard system configuration of the APWR and current PWR is shown in table 3.1.

2.2.5 Performance Verification

The Japanese PWR utilities and Mitsubishi Heavy Industry carried out a performance test program on the Advanced Accumulator from 1994 to 1996 to confirm the flow switching mechanism of vortex damper and to obtain flow performance data simulating LOCA condition. Figure 3.5 shows the test device for flow performance test simulating LOCA condition. The test accumulator was sized as vertical full scale and radial 1/2 scale model.

Table 3.1 APWR Engineered Safeguard System

	APWR	Current 4-loop PWR
Mechanical Subsystems	50% x 4	100% x 2
■ Accumulator	33% x 4 (Advanced Type)	33% x 4
■ Safety Injection System		
High Head Injection Pump	50% x 4	100% x 2
Low Head Injection Pump	Ø eliminated by adoption of the advanced accumulator	100% x 2 (Common with RHRS)
■ Containment Spray System	50% x 4 (Common with RHRS)	100% x 2
■ Emergency Water Storage	Inside Containment	Outside Containment
Electrical Subsystems	100% x 2	100% x 2

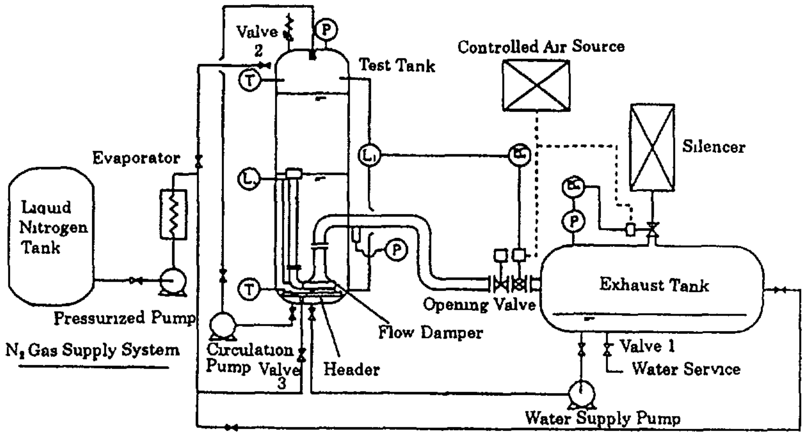


FIG. 3.5 Advanced Accumulator Performance Test Device

In this test program for the advanced accumulator, following results were obtained.

- ♦ Flow change mechanism inside flow damper was visually confirmed.
- ♦ Flow switch from large flow to small flow was confirmed to be smoothly performed.
- ♦ Similarity law of the vortex damper was confirmed using 1/2 and 1/5 scale models.
- ♦ Flow injection characteristic data was obtained to apply for actual accumulator design and licensing approval with establishment of analysis codes.

2.2.6 Design Evaluation

An extensive design, analysis and testing were conducted for the advanced accumulator. The design evaluations are as follows.

- ♦ A highly reliable ECCS system design can be obtained with extension of the safety function of the accumulator that is categorized as a passive equipment. The advanced accumulator contributes to the safety enhancement of the APWR, together with the in-containment RWSP and mechanical 4 subsystems.
- ♦ System design simplification can be achieved by elimination of conventional low head injection pumps.

3 CONCLUSION

In the APWR design, improvements in terms of safety, reliability and economy have been incorporated based on comprehensive evaluations of both the advanced technologies available today, and the experience associated with construction and operation of current PWR plants.

The radial reflector and the advanced accumulator are representative features of the APWR, which contribute to the enhancement of safety and reliability of the plant, and to the improvement of the plant's economics. These newly introduced technologies have been verified thorough extensive tests by Japanese industry groups. Therefore, it is concluded that these features comply with Japanese utilities' requirements.

IAEA-SM-353/23

MAJOR NSSS DESIGN FEATURES OF THE KOREAN NEXT GENERATION REACTOR

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Abstract

In order to meet national needs for increasing electric power generation in the Republic of Korea in the 2000s, the Korean nuclear development group (KNDG) is developing a standardized evolutionary advanced light water reactor 9alwr), THE Korean Next Generation Reactor (KNGR). It is an advanced version of the successful Korean Standard Nuclear Power Plant (KSNP) design, which meets utility needs for safety enhancement, performance improvement and ease of operation and maintenance. The KNGR design starts fro the proven design concept of the currently operating KSNPs with uprated power and advanced design features required by the utility. The KNGR design is currently in the final stage of the basic design, and the paper describes the major nuclear steam supply system (NSSS) design features of the KNGR together with introduction of the KNGR development program.

1 INTRODUCTION

1.1 Background

The KNGR is an evolutionary ALWR being developed by the KNDG. The KNGR development project is a long-term governmental R&D program and is to complete an advanced and upgraded standard nuclear power plant (NPP) design using the KSNP design [1, 2] as a starting point. It is also to solve the Korean specific problems in securing sites for NPPs and to reduce the construction cost per unit electricity output in accordance with the economy of scale.

The KNGR development project consists of four phases. During the first phase ended in 1994, major effort was focused on finding the most suitable reactor type in Korea and to develop utility requirements for the selected type. Revolutionary and evolutionary types of the ALWRs were carefully studied by the KNDG, and as a result the evolutionary type was selected. The second phase that is currently under way and to be ended in February 1999 aims at completing a basic design of the KNGR which meets the utility requirements. Third phase, scheduled to be continued for three years after the second phase, is to focus on further optimization of the KNGR design including technical and economical improvements. Standard safety analysis report (SSAR) of the KNGR will be submitted to the regulatory body during this phase to obtain a design approval of the KNGR standard design in accordance with the one-step licensing procedure which is expected to be legislated in due time. Next phase is the construc-tion phase during which the detailed design will be performed to support the construction. According to the mid- and long-term construction plan of power plants in Korea, the first KNGR unit is scheduled to be put into the grid in 2010.

The KNGR design is being performed by a multi-disciplinary team of Korean professional engineers capable of incorporating features that enhance operability and maintainability to the benefit of the plant owner with their experiences in the design of the KSNP units.

1.2 Design Requirements

The KNGR design will incorporate advanced features to enhance safety, to increase margins, to improve operability and maintainability, and to reduce cost. The major design requirements, which are consistent with the top-tier requirements of the Korean Utility Requirements Documents (KURD) [3], are shown in Table 1 along with those of the KSNP for comparison purposes.

Table 1. Comparison of Major Design Requirements for KNGR and KSNP

Items	KNGR	KSNP
Capacity	4000MWt	2825MWt
Plant design lifetime	60 years	40 years
Seismic design	SSE 0.3g	SSE 0.2g
Safety requirements		
- Core damage frequency	$< 10^{-5}$/RY	$< 10^{-4}$/RY
- Containment failure frequency	$< 10^{-6}$/RY	$< 10^{-5}$/RY
- Occupational radiation exposure	< 1 man·Sv/RY	< 1.2 man·Sv/RY
- Operator action time	Min. 30 minutes	Min. 10 minutes
- SBO coping time	Min. 8 hours	Min. 4 hours
- Thermal margin	10-15%	8%
- Hot-leg temperature	$< T_{KSNP}$	327.3 °C
- Emergency core cooling system	4-train	2-train
	Direct vessel injection	Cold-leg injection
	IRWST	Outside RWT
Performance requirements		
- Plant availability	90%	87%
- Unplanned trip	< 0.8/year	< 1/year
- Refueling cycle	18-24 months	15-18 months

The thermal power of the KNGR, 4000 MWt, has been increased by approximately 40% from the current 2825 MWt KSNP. With the power increase for the KNGR, preliminary size modification of the primary components has been completed during the basic design phase providing the required thermal margin throughout the 60-year plant design life as recommended in the KURD.

1.3 Development Procedure

The KNGR NSSS has evolved from the proven design of the currently operating KSNPs. Figure 1 depicts the KNGR development procedure. With the utility requirements recommended in the KURD, careful

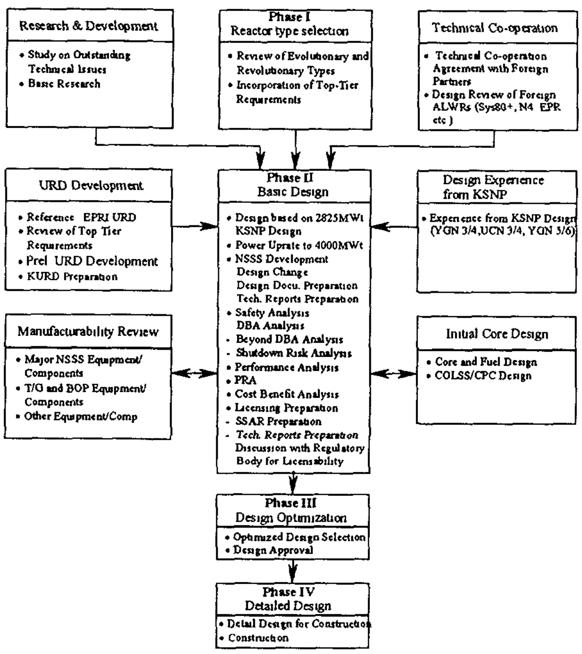

FIG. 1 KNGR Development Procedure

consideration has been given to the R&D results performed by universities and research institutes and information on foreign ALWRs including the advanced features. Also, the experiences in fabrication, construction, start-up and operating of the KSNPs have been reflected to the KNGR development. An intimate technical cooperation has been collaborated with the foreign partners from the beginning stage of the development. During the basic design phase, the analyses for design basis accidents (DBAs), beyond DBAs and shutdown risk, etc. have been performed as the design progresses and the results have been fed back to the KNGR design. Manufacturability of the major NSSS components and equipment has also been reviewed to secure the possible hardware provision.

During the design process, each addition or modification from the KSNP design has been assessed for its impact on safety, performance, operability, maintainability, and cost. Safety and performance analyses, probabilistic risk assessment (PRA) and cost-benefit analyses are used to assist in this evaluation process. Design results are to be evaluated to confirm the fulfillment of the requirements and iteration is to be continued either by re-performing the design process or modifying the requirements until the final design fix.

2 MAJOR NSSS DESIGN FEATURES

Development of the KNGR NSSS design has focused on reducing the hot-leg temperature to improve the safety margins of the reactor core, increasing the pressurizer volume to accommodate transients and reduce unnecessary challenges to the plant safety systems, and using improved material to reduce stress corrosion cracking of steam generator tubes, etc. The KNGR safeguards system includes safety injection system (SIS) with several advanced features such as four independent trains and direct vessel injection (DVI), in-containment refueling water storage tank (IRWST), and fluidic device in the safety injection tanks (SITs) to improve operability and to increase redundancy over the KSNP. The control room complex utilizes a great number of soft controllers and workstations emphasizing the human factors in designing the man-machine interface systems, thus providing far enhanced operability of the KNGR. The reactor vessel (RV) upper head area has been simplified to improve maintainability. The features that contribute to the improvements are summarized below.

2.1 Reactor Coolant System (RCS)

A schematic of the KNGR RCS with its major improvements is shown in Figure 2.

The RCS consists of a RV, two independent coolant loops connected to the reactor, and safety and auxiliary systems. Each loop consists of a 1066.8 mm ID outlet pipe, two 762 mm ID inlet pipes, a steam

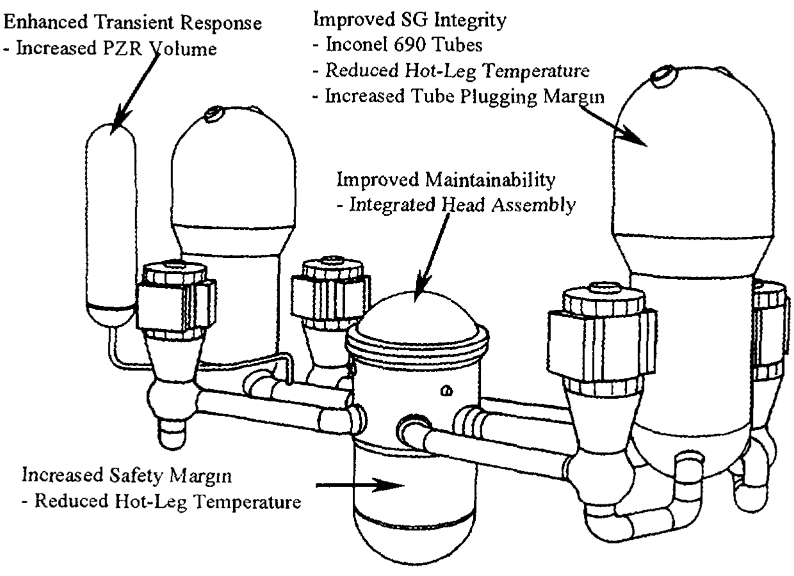

FIG. 2 Schematic of the KNGR Reactor Coolant System

generator (SG) and two reactor coolant pumps (RCPs) The RCPs are electric motor-driven single stage centrifugal pumps The RCS operates at a nominal pressure of 158 kg/cm^2, the system pressure is maintained by an electrically heated pressurizer (PZR) that is connected to one of the loops Core outlet temperature is 323 9 °C, which has been lowered by about 3 5 °C from the current KSNP design to provide additional core thermal margin, which allows more flexibility in operation It also increases margin against SG primary side corrosion attack The PZR volume has been increased by 33% to better accommodate transients and reduce unnecessary challenges to the plant safety system The SG incorporates design enhancements including the use of Inconel 690 tubes which dramatically reduces stress corrosion cracking, and 10% tube plugging margin to assure the capability to produce the design steam flow rate and pressure over entire design life with a significant number of tubes plugged Implementation of the pilot-operated safety relief valves (POSRVs) provides another advantage to the KNGR. Conventional spring-loaded safety valves connected to the PZR are replaced by the POSRVs, and functions of the RCS overpressure protection and safety depressurization could be performed by the POSRVs

2.2 Safeguards System

Comparing with the KSNP design, the most significant change of the KNGR NSSS is in the safeguards system design The SIS has been improved to provide a simpler and more reliable system with increased redundancy (Figure 3) Redundancy of the SIS is increased by having four independent mechanical trains The SIS takes suction from the IRWST and discharges directly into the downcomer of the RV The injected coolant flows directly into the RV to provide a simpler and more reliable system that avoids the potential of coolant loss in case of a cold-leg break accident inherent in the previous cold-leg injection scheme The common headers and associated valves are eliminated, and the low-pressure safety injection function and the switchover from the external water supply to the containment sump are also eliminated This SIS configuration improvement is a major contributor in reducing the core damage frequency by one order of magnitude from the KSNP design, according to the preliminary PRA results

Safety injection and containment spray pumps take suction from a storage tank, IRWST, located low inside the containment completely surrounding the reactor cavity The ability to cope with a degraded core accident has been enhanced by providing an in-containment water source to flood the cavity A flow regulating device, a fluidic device, is being considered to be installed in the SITs It takes an advantage of utilizing water inventory in the SITs by controlling the injected flow rate, and thus minimizes the possible injected coolant loss in the event of large break loss-of-coolant accident (LOCA) The performance tests are currently being conducted and a decision will be made for its implementation during the design optimization period

For reconfiguration of the SIS, exclusive studies have been performed Preliminary safety analysis has been carried out to confirm whether the results with the determined parameters and selected components

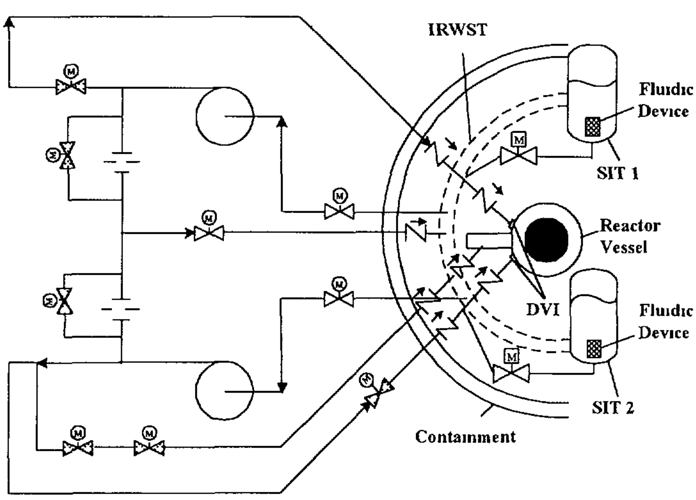

FIG 3 KNGR Safeguards System

sizing to meet the acceptance criteria, and the results have been fed back to the SIS design. For the DVI, several candidate nozzle locations are considered and being reviewed with respect to the PTS concern as well as its manufacturability.

Another design improvement is the interconnection of the shutdown cooling system (SCS) and the containment spray system (CSS), which allows the pumps of each system to serve as a backup for the other. Operation of the SCS is now simplified by eliminating the need to shift alignment from low-pressure safety injection to shutdown cooling.

2.3 Auxiliary Systems

The chemical and volume control system (CVCS) of the KNGR has been improved, especially in letdown and charging systems. The letdown flow control devices are located downstream of the letdown heat exchanger, which allows for an increased life and reduced maintenance of the letdown valves and orifices resulting from subcooling of the letdown flow prior to pressure reduction. The letdown heat exchanger is located inside containment, which minimizes high energy piping runs outside containment and is beneficial with respect to the ALARA principle. Three positive displacement pumps of the KSNP have been replaced to two centrifugal charging pumps each with flow capacity equal to the system design flow, which results in significantly less maintenance requirements and better reliability.

The KNGR is designed to reduce the level of pressure challenges to all systems interfacing with the RCS. General design features to address intersystem LOCA challenges consist of an increase of the system design pressure and incorporation of design features which terminate and/or limit the event by means of isolation or pressure relief.

2.4 Instrumentation and Control (I&C) Systems

The I&C systems of the KNGR have been designed to meet all the relevant requirements recommended by the KURD, which emphasizes human factor engineering in designing the advanced man-machine interface systems (MMIS). The I&C systems fully support the advanced control room design and, are characterized by state-of-the-art technologies, such as plant wide data communication network (DCN),

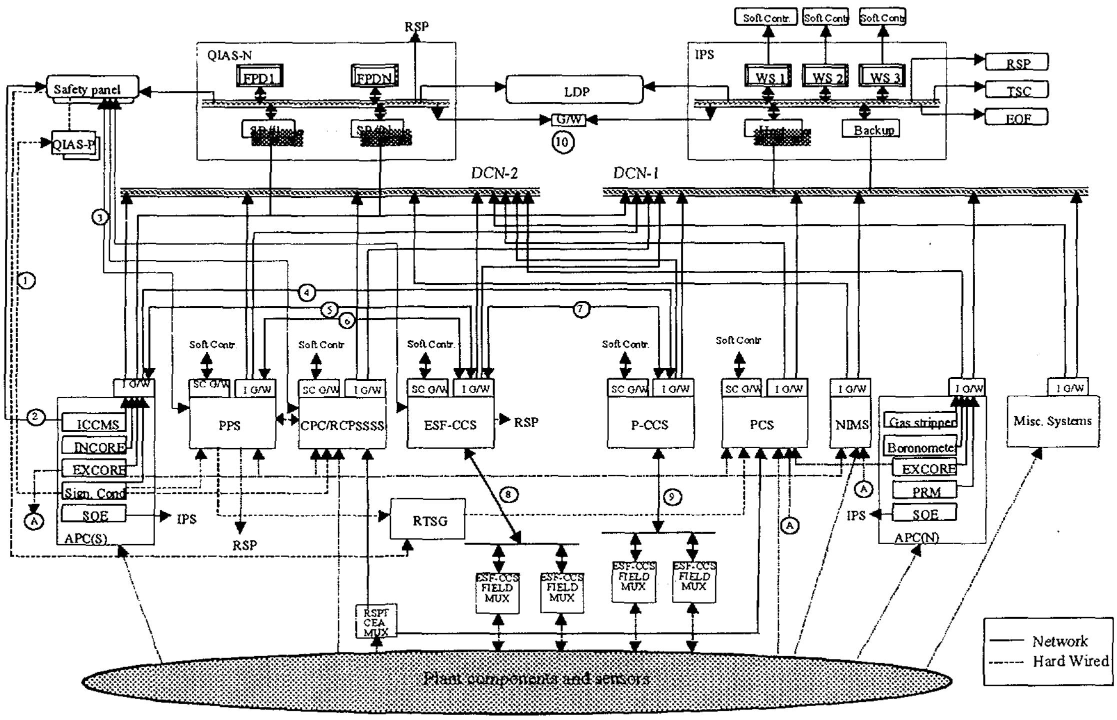

FIG. 4 I&C Overview Diagram of KNGR

distributed digital processing, advanced alarm and display processing, enhanced operator aid functions, signal multiplexing and soft control concept, etc. They utilize a top-down design concept in designing to achieve full integration with the various MMIS of the main control room (MCR). The MCR includes compact work stations consisting of two sets of operator's work stations and one additional set of backup work station, a large display panel (LDP), and the qualified indication and alarm system (QIAS). These advanced features replace the conventional display and controls used in the KSNP design and therefore, enhance operability of the KNGR. Figure 4 shows an overview of the KNGR I&C systems.

2.5 Integrated Head Assembly

The RV upper head area of the KSNP consists of many components, which are usually disassembled, separately stored and reassembled during every refueling outage. In order to make ease on this undesirable procedure and to simplify the complicated upper head region structure, the concept of an integrated head assembly (IHA) is adopted in the KNGR (Figure 5).

The IHA contains not only the RV upper head, control element driving mechanisms (CEDMs), heat junction thermocouples and head lift rig as in the KSNP, but also the head area cable tray, missile shield, seismic restraints and the CEDM cooling fan and ducts, which now can be handled together as a package. It is designed compact by making the CEDM cooling air flow inside the enclosing shroud and locating the CEDM cooling fans on the missile shield plate. The main columns and the lifting frame are designed to satisfy the NUREG-0612 [4] requirements and each component is designed to be shipped and installed easily. The IHA contributes to the reduction in radiation exposures to the installers, as well as the reduction in refueling outage duration.

3 CONCLUSION

The KNGR development is a national long-term R&D program consisting of four phases. The reactor type was selected and utility requirements were developed during the first phase. Basic design of the KNGR is under way in the second phase, and a design optimization process will follow before the finalization of the KNGR design configuration in the third phase. Construction of the first KNGR unit as a final product of the fourth phase through the detailed design work is planned to be completed in 2010.

The KNGR is a standardized evolutionary ALWR upgraded from the proven KSNP design. Compared with the KSNP, which serves as a starting point of this development, the KNGR NSSS incorporates several

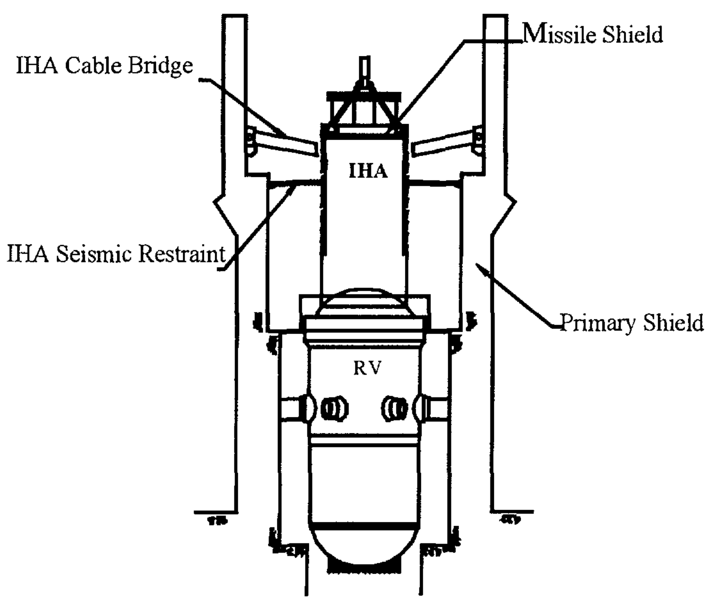

FIG. 5 KNGR Integrated Head Assembly

advanced design features, which may lead to safety enhancement and performance improvement as described in this paper.

With the current KNGR design, more safety margin is expected and the results of the PRA show one order of magnitude reduction in core damage frequency compared with those of the KSNP. This low risk of core damage compares favorably with the figure suggested by the IAEA [5] for "Good Plants" in the year of 2000. Such high degree of the KNGR design improvement assures a high degree of certainty in its successful application to the future plants required by Korea and moreover by the world.

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IAEA-SM-353/25

XA0053554

KEY DEVELOPMENTS OF THE EP1000 DESIGN

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Abstract

In 1994, a group of European utilities initiated, together with Westinghouse and its industrial partner GENESI (an Italian consortium including ANSALDO and FIAT), a program designated EPP (European Passive Plant) to evaluate Westinghouse passive nuclear plant technology for application in Europe. The Phase 1 of the European Passive Plant program involved the evaluation of the Westinghouse 600 MWe AP600 and 1000 MWe Simplified Pressurized Water Reactor (SPWR) designs against the European Utility Requirements (EUR), and when necessary, the investigation of possible modifications to achieve compliance with the EUR. In Phase 1 of the program, which has been completed in 1996, the following major tasks were accomplished: The impacts of the European Utility Requirements (EUR) on the Westinghouse nuclear island design were evaluated. A 1000 MWe passive plant reference design (EP1000) was developed which conforms to the EUR and is expected to be licensable in Europe. With respect to the NSSS and containment, the EP1000 reference design closely follows those of the Westinghouse SPWR design, while the AP600 design has been taken as the basis for the design of the auxiliary systems. Extensive design and testing efforts have been made for the AP600 and SPWR during the respective multi-year programs. While the results of these programs have been and will continue to be utilised, at the maximum extent, to minimise the work to be performed on the EP1000 design, the compliance with EUR is a key design requirement for the EP1000. The ultimate objective of Phase 2 of the program is to develop design details and perform supporting analyses to produce a Safety Case Report (SCR) for submittal to European Safety Authorities. The first part of Phase 2, hereafter referred as Phase 2A, started at the beginning of 1997 and will be completed at the end of 1998. Scope of this phase of the program is to develop the design modifications of important systems and structures so to comply with the EUR. This paper provides a brief description of the most significant developments of the EP1000 plant design during Phase 2A of the EPP program.

1. INTRODUCTION

In 1994, a group of European utilities initiated, together with Westinghouse and its industrial partner GENESI (an Italian consortium including ANSALDO and FIAT), a program designated EPP (European Passive Plant) to evaluate Westinghouse passive nuclear plant technology for application in Europe. The European utility group consists of the following organizations:

- Agrupación eléctrica para el Desarrollo Tecnologico Nuclear (DTN), Spain
- Electricité de France, France
- ENEL, SpA., Italy
- Imatran Voima Oy, Finland
- Scottish Nuclear Limited (acting for itself and on behalf of Nuclear Electric plc), U.K.
- Tractebel Energy Engineering, Belgium
- UAK (Represented by NOK-Beznau), Switzerland
- Vattenfall AB, Ringhals, Sweden

(*) This paper is presented on behalf of the EPP-SC and EPP-TG. Mr. Noviello is the Chairman of the EPP-SC. Special thanks go to Mr. Saiu of Ansaldo for his extensive contribution to the paper.

The Phase 1 of the European Passive Plant program involved the evaluation of the Westinghouse 600 MWe AP-600 and 1000 MWe Simplified Pressurized Water Reactor (SPWR) [1] designs against the European Utility Requirements (EUR) [2] and, when necessary, investigation of possible modifications to achieve compliance with the EUR.

The base design for these passive plant programs is the AP600. The passive PWR approach to design is to strike a balance between proven technology and new concepts - the advantage of the traditional Westinghouse two-loop PWR combined with natural circulation passive safety systems. The result is a greatly streamlined plant that exceeds safety regulations and availability requirements, is economically competitive and promote broader public confidence in nuclear energy.

With respect to safety systems and containment, the EPP design closely follows that of the Westinghouse SPWR design, while the AP600 plant design has been taken as the basis for the auxiliary systems. However, the EP1000 design also includes features required to meet the EUR, as *well as key European licensing requirements* [3].

The ultimate objective of Phase 2 of the program is to develop design details and perform supporting analyses to produce a Safety Case Report for submittal to European Safety Authorities. The first part of Phase 2, hereafter referred as Phase 2A, started at the beginning of 1997 and will be completed at the end of 1998. Scope of this phase of the program is to focus on improving the design of important systems and structures.

In parallel to the Phase 2A effort, a group of European Utilities are sponsoring the activities for the preparation of the EP1000 EUR Volume 3. Volume 3 of the EUR is intended to the compliance against the EUR. The EP1000 EUR Volume 3 program began in June 1997 and will be concluded at the end of 1998.

In the following, the most significant technological developments of the EP1000 plant design during Phase 2A of the EPP program are described briefly.

2. NUCLEAR SYSTEM DESIGN

Phase 2A activities have focused on improving the design of important systems and structures, including: reactor coolant system (Fig. 1), reactor safety systems and major auxiliary systems and to perform the safety analyses to support system design. Several minor modifications have been implemented in each one of the above. In the following some of the most important changes implemented in Phase 2A will be discussed.

2.1. Reactor Vessel and Core design

The core, reactor vessel, and reactor internals of the EP 1000 (Refs [6 -7]), are similar to those of currently operating Westinghouse PWR plants, but several new features are incorporated to enhance the performance characteristics as compared with existing plants.

The reactor core uses the Westinghouse 12 foot (3658 mm), 17x17 fuel assembly. A low-power density is achieved by making the core larger than previous 1000 MWe designs, with the number of fuel assemblies increased from 157 to 193. This configuration results in core power density and average linear power density reduction of about 25 percent, to 88,4 kW/l and 15,6 kW/m, over existing plants of the same power rating. This results in lower fuel enrichments, less reliance on burnable absorbers, and longer achievable operating cycles.

The core is surrounded by a stainless steel radial neutron reflector that contributes to lowering fuel cycle cost and to reduce neutron fluence on the reactor vessel wall, an important factor in view of the 60 year lifetime design objective.

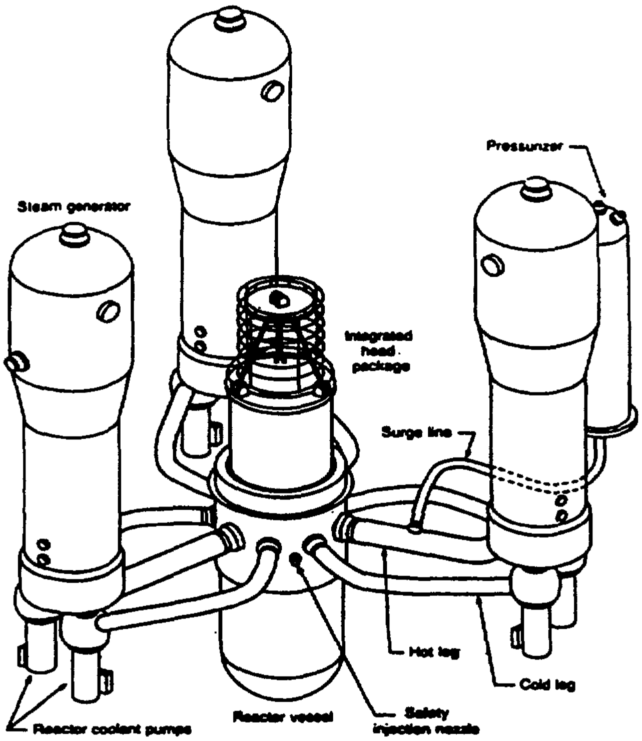

FIG. 1. EP 1000 Reactor coolant system layout

The EP1000 core is optimized for UO_2 fuel assemblies. However, provisions are made to allow the use of up to 50% standard MOX fuel assemblies in the core.

During Phase 2A, activities have been performed to design both UO_2 and 50% MOX core that could meet the EUR. The results of UO_2 Core Design activities will be highlighted in the following.

Two preliminary 24-months cycle UO_2 fuel management schemes have been developed for the EP1000 reactor as part of the Phase 2A Program. These schemes are required to meet the EUR Revision B Low Boron Design Requirements, as well as all applicable conventional safety analysis and design limits. The low boron capability sets out to reduce boron dilution risks and to provide an adequately negative moderator feedback to avoid damage of the core and of the RCS pressure boundary during any ATWS through 100% core life and finally to reduce ALARA costs through improvement of the chemistry.

One core design utilises mid-enriched (2,0 weight percent [w/o]) axial blankets that provide an economic benefit equivalent to approximately a 0.1 w/o reduction in fuel average enrichment, when compared to the second design which utilises a single uniform axial fuel enrichment. The only observed consequence resulting from the use of these mid-enriched axial blankets is an acceptable increase of base load steady state axial peaking factor (F_Z), and of the nuclear heat flux hot channel factor, (F_Q). The nuclear enthalpy rise hot channel factor, $F_{\Delta H}$, during steady state operation is, however, comparable for both designs. Both designs assume that a stainless steel radial reflector (similar to the one utilised for the AP600 and SPWR) is employed. For the initial core design, discrete Wet Annular Burnable Absorbers (WABA) rodlets, which are consolidated into burnable absorber assemblies, and Integral Fuel Burnable Absorbers (IFBAs) are used. Discrete absorber designs, integral fuel burnable absorber designs, or both, may be used in subsequent reloads. (IFBAs are the preferred burnable absorber design).

Another core design feature is the use of reduced-worth control rods (termed "gray" rods) to achieve daily load-follow capability without requiring daily changes in the soluble boron concentration (see IAEA-TECDOC-968).

2.2. Reactor Vessel Internals

The reactor vessel internals design has been reviewed during Phase 2A as a consequence of the requirements related to Design Extension Conditions (DEC) of the EUR. The ability of the EP1000 to provide in-vessel retention and cooling of core debris following DEC was evaluated as part of the In-Vessel Debris Retention Study. The study identified a problem with in-vessel coolability and the need for a modification to the reactor vessel lower internals configuration to solve the issue. The bottom of the EP1000 lower core support plate sits higher in the reactor vessel than in the reference plant design (AP600) and the additional metal mass of the support plate and reflector would not be submerged into the molten metal pool. The resulting melt geometry produces a heat flux profile that exceeds the critical heat flux for the reactor vessel and as a consequence the reactor vessel would fail. A revised configuration, that places the bottom of the core support plate approximately 10 inches (260 mm) lower, allows contact between the lower support plate and the molten debris pool. The increased thickness of the molten metal layer spreads the heat over a larger area of the reactor vessel and reduces the heat flux below the critical heat flux. This design change to the lower internals should be further evaluated as part of the Phase 2B design activities.

2.3. Auxiliary Cooling Systems

The EP1000 Auxiliary Cooling Systems include:

- Normal Residual Heat Removal System (RNS)
- Component Cooling Water System (CCS)
- Service Water System (SWS)
- Spent Fuel Pool Cooling Water System (SFS)

Activities have been performed in Phase 2A to integrate the design of the above systems so to provide a configuration that meets the European Utility Requirements. In particular, the design activities have focused the heat transfer chain composed by RNS, CCS and SWS.

While the systems still retain the same configuration of the SPWR reference plant, several modifications have been made driven by EUR requirements.

The Normal Residual Heat Removal System (RNS) consists of two separate mechanical trains of equipment. Each train consists of one residual heat removal (RHR) pump and one heat exchanger. In addition, RNS comprises piping, valves and instrumentation necessary for correct system operation. The RNS is located completely inside the containment. The RNS is designed to perform its functions in a very reliable and failure tolerant manner. The reliability is achieved with the use of highly reliable and redundant equipment and with a simplified design. According to the EUR, the RNS is a F2 safety system, designed in accordance with Equipment Class D Standards since it provides defence-in-depth functions that contribute to the overall safety of the plant.

The European Utility Requirements (use of MOX fuel, Boron Recycling, Cooldown time limits and site conditions) directly affect the design of the RNS [3]. The Heat Removal Design Bases set by the European Utilities Requirements are the following:

- The Plant should be capable of shutdown from Hot Zero Power to cold shutdown at a temperature less than 60 °C (140 °F) within 20 hrs.

- Initiation of RNS operation 6 hrs after reactor shutdown.
- Maintain RCS T < 60 °C (140 °F) during refuelling with one train unavailable, beyond a time period from reactor shutdown compatible with availability targets.
- Reduce RCS T at 180 °C (356 °F) during Hot Shutdown with one train available, within 12 hrs after shutdown.
- Bring the RCS temperature to 90 °C (194 °F) within 36 hrs after shutdown with a single failure in the RNS system.

The heat load data have been calculated taking into account the EUR. In particular, in:

- MOX decay heat
- Spent Fuel Pool Cooling System takes into account the increased storage capacity requirements and MOX decay heat
- Impact of requirement to recycle boron and the additional waste evaporator heat loads.
- In addition, heat Loads during refuelling are impacted by the very aggressive EP1000 refuelling schedule that requires full core off-load in about 108 hours to be able to complete a refuelling in 13 days.

Different options, in terms of heat exchanger sizing and systems flow rates, have been evaluated to meet these design requirements. The final design includes larger CCS heat exchangers, such that a lower temperature can be reached at the RNS heat exchanger inlet and so that the RNS heat exchangers size increase can be limited.

Moreover, to increase heat transfer effectiveness an RNS two-shell pass heat exchanger design has been utilized in place of the one-shell pass design of the reference plant (SPWR).

3. CONTAINMENT DESIGN AND LEAKTIGHTNESS

Among all the EUR requirements to be considered in the EP1000 plant design, there is a group that challenges the containment design beyond current practice. These requirements are related to consideration of Design Extension Conditions (selected Severe Accident Conditions and Complex sequences) in the plant design. It must be ensured that the containment leak-tightness is maintained for the duration of the accident, up to the accident termination.

The overall issue of the Containment Leaktightness is addressed by means of three main defense barriers:

- Containment Isolation
- Primary Containment Leak-tightness
- Secondary Containment

3.1. Containment Isolation

The EP1000 containment isolation is significantly improved over that of conventional PWRs. One major improvement is the large reduction in the number of penetrations. Furthermore, the number of normally open penetrations is reduced by 60 percent (Refs [2-3]). For example, the chemical and volume control system (CVS) letdown penetration is normally closed because CVS purification is performed in a high-pressure loop inside containment. Also, there are no penetrations required to support post-accident mitigation functions (the canned motor reactor coolant pumps do

TABLE I. CONTAINMENT ISOLATION/BYPASS SIMPLIFICATION

Component	Conventional 3-Loop Plant	EP1000
Penetrations	93	50
Normally Open Penetrations	38	15

not require seal injection, and the residual heat removal and safety injection features are located entirely inside containment). Table I provides a summary of the EP1000 penetrations.

3.2. Primary Containment Leak-tightness

Primary containment leak-tightness depends on the behaviour of the different components that constitute the containment pressure boundary barrier, in particular:

- The primary containment shell, including the basemat, which envelops the reactor vessel, the primary system and most or all systems containing highly radioactive fluids, after a fuel damage accident.
- Mechanical penetrations which include the mechanical piping penetrations (i.e., the transitions between pipes penetrating the containment boundary and the boundary itself) and the fuel transfer tube.
- Equipment hatches
- Airlocks
- Electrical penetrations which permit the penetration of power and control cables
- Isolation valves of pipes and ducts which cross the containment boundary

3.2.1. Containment Vessel

The Containment Vessel is a free-standing cylindrical steel vessel with elliptical upper and lower heads.

During Phase 2A, detailed design activities have been performed to define containment configuration. The design of the steel containment structure has been performed according applicable ASME, ASTM and AISC standards.

The evaluations have been performed modelling the containment vessel with a three-dimensional finite element model. The Containment shell has been modelled with thin shell element while the Polar Crane and the horizontal stiffeners with beam elements. Discontinuities that include hatches and main penetrations have been accounted for.

The structures have been checked at ASME service level C. It is assumed that exceeding this limit will result in a loss of leak-tightness.

The analysis has shown that with an internal pressure of 1 MPa (10 bar) and with a temperature equal to 204 °C, the Service Level C is still meet.

The location at which the stress limits are reached is in the Main Steam Penetration area, while about 20% margin still exist in the Equipment Hatch area.

Ultimate failure pressure is expected to be significantly higher.

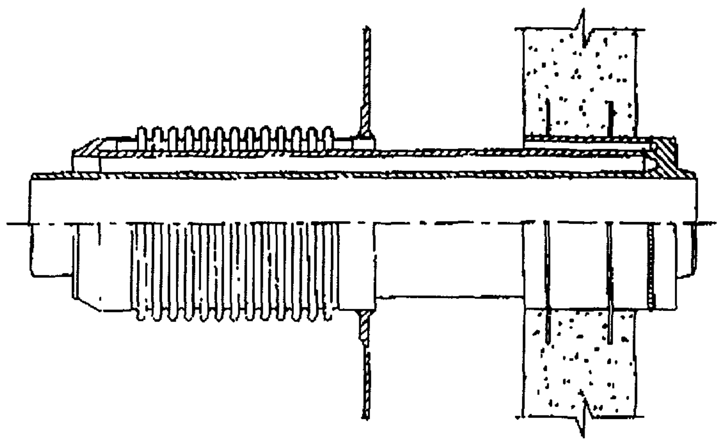

FIG. 2. EP1000 Main Steam Line Penetration

3.2.2. Critical Analysis of Reference Design

The analysis of the Reference Designs (i.e., SPWR and AP600) has identified the following critical areas for which leak-tightness improvements can be realized:

- the most critical components are the large containment penetrations (personnel airlock and equipment hatches) mainly because their leak-tightness depends on elastomeric materials which are subject to thermal and radiation damage and ageing;
- the design of high-energy mechanical penetrations (i.e, Main Steam and Feedwater lines) has to be improved in order to consider the potential effects of the increased challenges deriving from Design Extension Conditions.

Moreover, further evaluations are needed for electrical penetrations where the sealing area arrangement can be challenged by temperature effects.

3.2.3. Leak-tightness Improvement Measures

Review of the reference design penetration resulted in a different mechanical penetration design, as well as in a different global support distribution for the Main Steam and Feedwater Lines.

The suggested penetration design, Fig. 2, is characterized by a metallic bellows between the Main Steam/Feedwater line and the primary metallic containment shell in order to assure leak-tightness behaviour of the whole system, as well as to decouple the steam line from the metallic shell. In addition, the penetration anchorage has been moved from the Turbine Building boundary wall, as in the AP600, to the shield building wall. By providing the anchor point on the shield building wall, the bellows has only a leak-tightness function and cyclic loadings on the bellows, due to cyclic thermal expansion of both the MS/MF line, are minimised.

An improved Equipment Hatch Design has also been proposed. The design, developed by ENEL, Fig. 3, is characterised by two sealing areas that are connected through an annular space dedicated to collect and confine potential leakage through the inner gasket.

Heat dissipation mechanisms are improved by simple arrangements to limit the temperature at the outer seal below the value at which the gasket seal material has an unstable behaviour. The gasket configuration is such as to promote the metal to metal contact in the inner sealing area in case of a complete gasket degradation.

Finally, depending on the DEC temperatures which will be considered, a thermal insulation to shield the in containment equipment hatch portion against local thermal loads is foreseen.

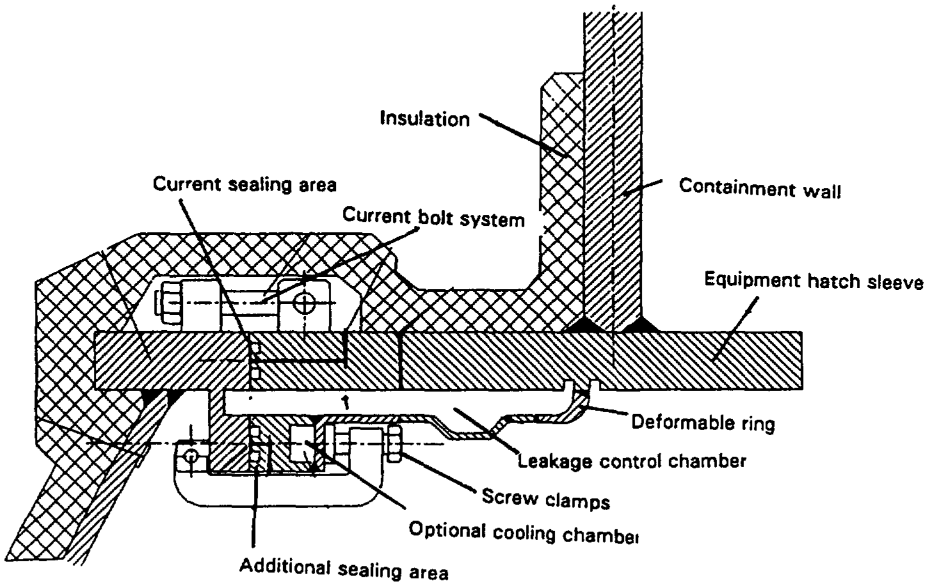

FIG. 3 - EP1000 Improved Airlock

Similar to the equipment hatch, the personnel airlock features have been studied to limit the concern related to temperature and radiation damage of the gasket elastomeric material. A full scale test program [6], (ATHERMIP test facility, designed by ANSALDO) has been conducted, outside of the EPP program by ENEL (Italy) with the participation of EDF (France), Empresarios Agrupados (Spain), and the University of Pisa (Italy) to evaluate the effectiveness of the proposed solution and qualify the solution for DEC conditions.

3.2.4. Secondary Containment Ventilation System (PAFS)

The EP 1000 is equipped with a secondary containment. For the steel containment reference configuration, the secondary containment is defined as the structure which confines the penetration area (annulus) to collect leakages through the penetrations that constitute the major source of containment leakage. The Secondary Containment Ventilation System (Passive Annulus Filtration Systems - PAFS) is part of the EP 1000 HVAC systems. It is designed to collect and filter the leakages through the penetrations to limit the offsite dose following a severe accident.

The PAFS is designed to perform the following major functions:

- Contribute to the limitation of the offsite dose to the value defined by site parameters; and
- Maintain a negative pressure in the annulus penetration (secondary containment).

Initial evaluations indicate the EP1000 radioactive releases to the environment will be low, in compliance with the EUR safety targets for Design Basis Accidents (DBA) without operation of the PAFS. Therefore, the PAFS is provided to fulfill a level F2 safety function which in the EUR Rev. B is defined as, "to ensure that the releases are kept within the targets set for DEC (design extension conditions)".

The PAFS, Fig. 4, is connected to the middle and lower annulus of the secondary containment. It consists of two mechanical trains of equipment. Each train consists of one HEPA filter, one eductor and a compressed air storage. The motive force of the eductor is the compressed air stored in tanks, having a capacity per train to support the function of the PAFS for the first 24 hours after a DEC accident. The capacity of both trains, used one after the other, should be able to perform the PAFS function for a period of 72 h.

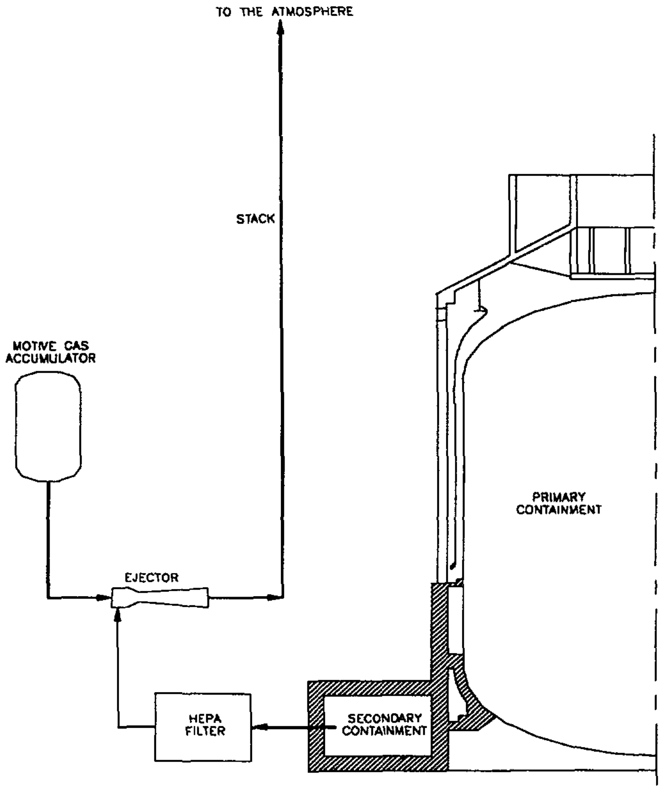

FIG. 4. Secondary Containment Ventilation System Simplified Sketch

4. SAFETY CONCEPTS

4.1. Safety requirements and design philosophy

The EP1000 safety philosophy is aimed at the prevention of accident but also gives attention to the mitigation of the consequences of accidents that could give rise to major releases. The aim is to reduce both the probability of the events and their associated off-site consequences in order to avoid the need for extensive countermeasures and to offer the authorities the possibility to simplify the off site emergency planning.

The basic EP 1000 safety philosophy is based on utilization of inherent margins (e.g. larger volumes and water inventory, lower power density, negative power and temperature reactivity coefficients) to limit system challenges.

Consistent with current practice, active systems are used as first level of defence against the most probable events.

The EP1000 uses, as a second line of defence, passive safety systems (Refs [6-7]). The use of passive safety systems has provided significant and measurable improvements in plant simplification, safety, reliability, and investment protection. These passive safety systems provide a major enhancement in plant safety and investment protection as compared with conventional plants. These systems provide reactivity control, establish and maintain core cooling and containment integrity, with no operator or AC power support requirements. The passive systems are designed to meet the U.S. NRC single-failure criterion, and probabilistic safety assessments (PSAs) are used to verify their reliability.

Finally, an additional level is called for, namely prevention and mitigation of Severe Accident Conditions through the consideration of Design Extension Conditions, formerly considered Beyond the Design Basis. The approach to Design Extension Conditions is reported in the following.

4.2. Design Extension Conditions

The assessment of the EP1000 performance against severe accidents is performed in agreement with European Utilities Requirements. The general approach to severe accidents identifies the sequences to be reduced in probability below the credibility threshold and those to be mitigated.

According to the EUR, the assessment of the Design Extension Conditions (DEC) in addition to the Design Basis Accident (DBA) is the preferred method for giving consideration to the complex sequences and severe accidents at the design stage without including them in the Design Basis Conditions.

The assessment of the DEC permits the definition and evaluation of the Design Extension Measures (DEM) to prevent core melting or mitigate the consequences of accident sequences such as:

- Complex sequences which involve failures beyond those considered in the deterministic Design Basis
- Severe accidents, both to prevent early and delayed containment failure and to minimize releases for the conditions that go beyond the Design Basis Conditions (DBC).

The Design Extension concept makes use of probabilistic methods (PSA) as one way to identify the need for the implementation of measures including upgraded or additional equipment or accident procedures for complex sequences and severe accidents.

A preliminary Probabilistic Safety Assessment shows that the EP1000 plant has a level of risk similar to AP600 and meets and exceeds the design goals specified by the EUR. Preliminary evaluations provide a core damage frequency of 8.3×10^{-8} per year for internal events at power conditions. In particular, the analyses have shown that many of the events that, in the past, were leading contributors to the risk of nuclear power plants, such as interfacing LOCAs, are not equally significant for the EP1000.

5. PLANT LAYOUT

During Phase 2A, plant layout activities have been performed limited to the Nuclear Island.

Criteria for plant layout have been defined according to the Westinghouse practice revised, where needed, to account for the European Utilities requirement. The process of generating the layout involved participation of the European Utilities as well as of the Industrial Partners.

A complete 3D model of the NI has been generated during Phase 2A. The level of details is such to include: the NI civil structures, the reactor system, Nuclear Fluid Systems and Auxiliary Systems both in the reactor building and auxiliary building. The main lines (i.e., main steam lines, feedwater lines, Automatic Depressurization lines) have already been routed while, only in Phase 2B, the smaller piping and main HVAC and electrical cable tray routing will be completed.

Representative general arrangement drawings of the containment building and Nuclear Island are shown in Figures 5 and 6.

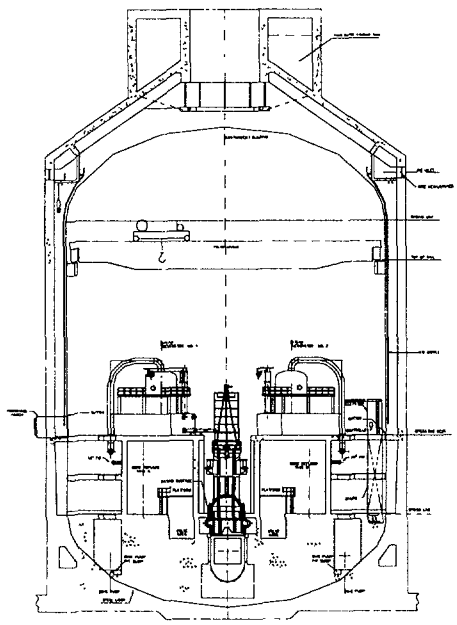

FIG. 5. General containment arrangement; elevation view at Section B-B

5.1. Containment Building

The containment vessel, a Seismic Category I structure, is a free standing steel cylinder, 46 meter in diameter and 67,6 meter height from the containment sump to the inside containment top head. It is surrounded by a Seismic Category I reinforced concrete shield building that provides protection against external events.

There are three floors (grade access, maintenance floor, and operating deck) and ten equipment compartments within the containment building. Floor gratings are provided for access to equipment at other elevations. The principal systems located within the containment building are the Reactor Coolant System (RCS), the Passive Core Cooling System (PXS), the Normal Residual Heat Removal System (RNS), and the Chemical and Volume Control System (CVS).

5.2. Auxiliary Building

The primary function of the auxiliary building is to provide protection and separation for the Seismic Category I mechanical and electrical equipment located outside the containment building. It also provides shielding for the radioactive equipment and piping that is housed within the building.

The auxiliary building is a Seismic Category I reinforced concrete structure that shares a common basemat with the containment building. The auxiliary building is a C-shaped section of the nuclear island that wraps around approximately 70 percent of the circumference of the shield building. Floor slabs and the structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

With respect to Phase 1 activities, the main differences in plant layout, are the definition of the Shield Building Roof and design of containment bottom.

The shield building roof conical design has been derived from the AP600. The SPWR shield building roof is semi-spherical to provide a more robust design because the Japanese Seismic Requirements (0.5 g Zero Peak Acceleration Design Basis Earthquake). The EP1000 is designed for a

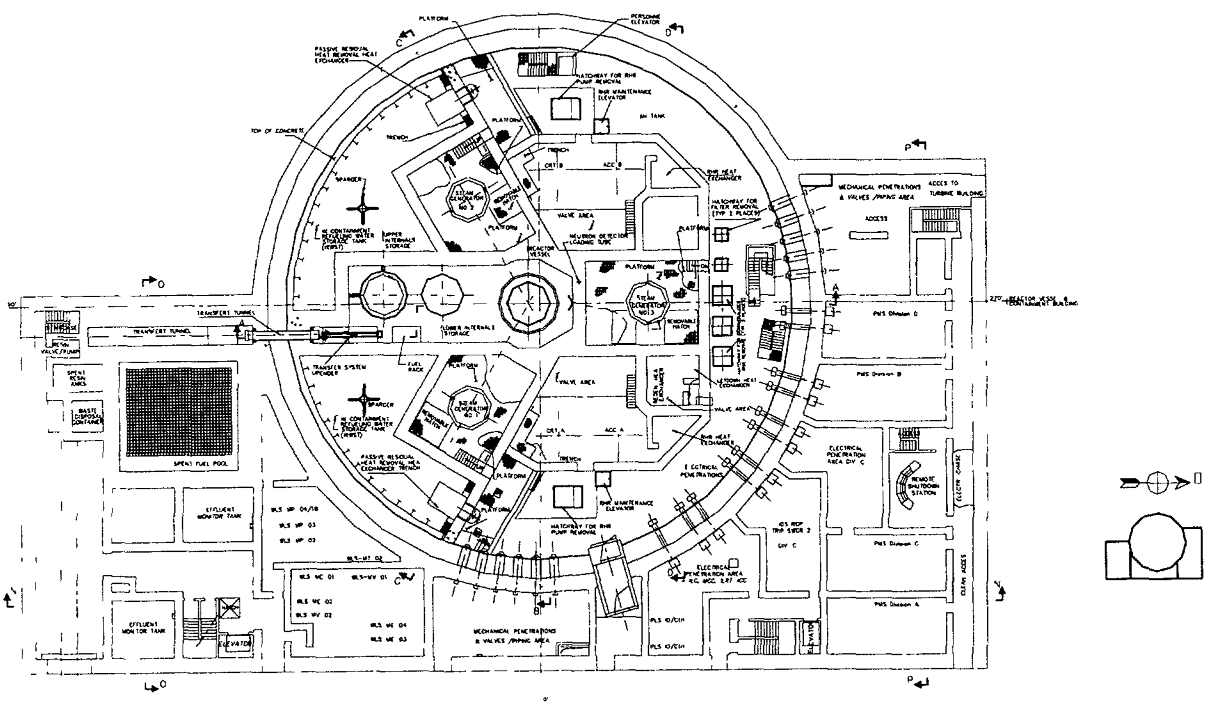

FIG. 6. Nuclear island arrangement; plan view at operating floor elevation

much lower Design Basis Earthquake hence the AP600 configuration has been chosen. This results in an important cost saving.

The containment bottom shape has been changed from a flat bottom, derived from the SPWR, to an elliptical bottom shape derived from the AP600 design. The reference SPWR configuration provides a flat bottom shape for the metallic containment; this configuration is not common for metallic containment characterized by high internal design pressure. The anchorage of containment vessel to the basemat is realized with massive anchorage that may affect the plant cost, the construction sequence and construction time schedule.

This change provides a more robust design since accidental pressure always acts as an internal force to the containment vessel. In addition, as a side advantage, the proposed solution facilitates the licensing process since the USNRC has approved the design for AP600.

6. PROJECT STATUS AND PLANNED SCHEDULE

The ultimate objective of Phase 2A of the EPP program is to develop design details and perform supporting analyses to produce a Safety Case Report for submittal to European Safety Authorities. The first part of Phase 2, "Phase 2A" is focusing on the definition and design of important systems and structures. Activities have already been performed both to define the design details of the important systems (e.g. Reactor Coolant System, Passive Injection and Core Cooling System, Passive Containment Cooling System, etc.), and to address some specific EUR requirements including Hazards (i.e., Aircraft Crash, Gas Cloud Explosion), Design Extension Conditions and performance requirements (e.g., MOX Fuel, Low Boron Core, etc.) and finally EUR specific Site Interface Requirements (i.e., Seismic Margins, Soil Characteristic and site environmental conditions).

In parallel to the Phase 2A effort, a group of European Utilities are sponsoring the activities for the preparation of the EP1000 EUR Volume 3. Volume 3 will be the EP1000 plant example and compliance assessment against the EUR. The EP1000 EUR Volume 3 program began in June 1997 and will be concluded at the end of 1998.

The evaluation of the EP1000 design against EUR has shown, to date, only minor non-compliances that are traced and will be solved in the next phase of the EPP program.

The second part of Phase 2, "Phase 2B", should start at the beginning of 1999 and will be completed in the 2001.

Phase 2B will include both the analyses and evaluations required to demonstrate the adequacy of the design, and the preparation of a Safety Case Report.

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BWR 90 — THE ABB ADVANCED BWR DESIGN

XA0053555

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Abstract

ABB has two evolutionary advanced light water reactors available today - the BWR 90 boiling water reactor and the System 80+ pressurised water reactor. The BWR 90 is based on the design, construction, commissioning and operation of the BWR 75 plants. The operation experience of the six plants of this advanced design has been very good. The average annual energy availability is above 90%, and the total power generation costs have been low. In the development of BWR 90 specific changes were introduced to the reference design, to adapt to technological progress, new safety requirements and to achieve cost savings. The thermal power rating of BWR 90 is 3800 MWth (providing a nominal 1374 MWe net), slightly higher than that of the reference plant. ABB Atom has taken advantage of margins gained using a new generation of its SVEA fuel to attain this power rating without major design modifications. The BWR 90 design was completed and offered to the TVO utility in Finland in 1991, as one of the contenders for the fifth Finnish nuclear power plant project. Thus, the design is available today for deployment in new plant projects. Utility views were incorporated through co-operation with the Finnish utility TVO, owner and operator of the two Olkiluoto plants of BWR 75 design. A review against the European Utility Requirement (EUR) set of requirements has been performed, since the design, in 1997, was selected by the EUR Steering Committee to be the first BWR to be evaluated against the EUR documents. The work is scheduled for completion in 1998. It will be the subject of an *"EUR Volume 3 Subset for BWR 90"* document. ABB is continuing its BWR development work with the "evolutionary" design BWR 90+. The primary design goal is to develop the BWR as a competitive option for the anticipated revival of the market for new nuclear plants beyond the turn of the century, as well as feeding ideas and inputs to the continuous modernisation efforts at operating plants. The development is performed by ABB Atom together with TVO. Swedish BWR operators have also joined the project.

1. INTRODUCTION

Today, ABB has a modern BWR design, the BWR 90, that has already been offered commercially. This design was selected by the European Utility Requirements (EUR) group to be reviewed for compliance with its set of requirements. ABB is continuing its BWR development work, however, with focus on the 21st century, on a new design called BWR 90+ that offers reduced costs and significant safety improvements. The work aims at providing an economical alternative based on evolutionary development of the earlier advanced BWR design. The design goal is a 1500 MWe plant that can be built in less than 1500 days.

A second purpose of the development activities is to provide input to improvements and modernisation of earlier generations of nuclear power plants.

2. DEVELOPMENT BASED ON SUCCESSFUL OPERATION EXPERIENCE

ABB Atom has a long tradition of plant and system development activities related to nuclear power plants. Its first BWR unit, Oskarshamn 1 nuclear power plant in Sweden was taken into operation in 1972, - developed and built without reliance on licenses. This design incorporated a number of advanced features such as a pre-stressed concrete containment, fine-motion control rods and a passive isolation condenser. Subsequent plants were designed much along the same lines as Oskarshamn 1, with step-wise improvements.

A major step forward came with the advanced BWR 75 design, which is characterised by use of internal recirculation pumps, fine-motion control rod drives, four independent and physically separated trains of engineered safety systems, and a pre-stressed slip-formed containment. Six nuclear power plants of this design are in operation in Sweden and Finland. The accumulated successful operation experience of these plants amounts to almost 100 reactor-years, and demonstrates the capability of being operated at high energy availability factors (figure 1).

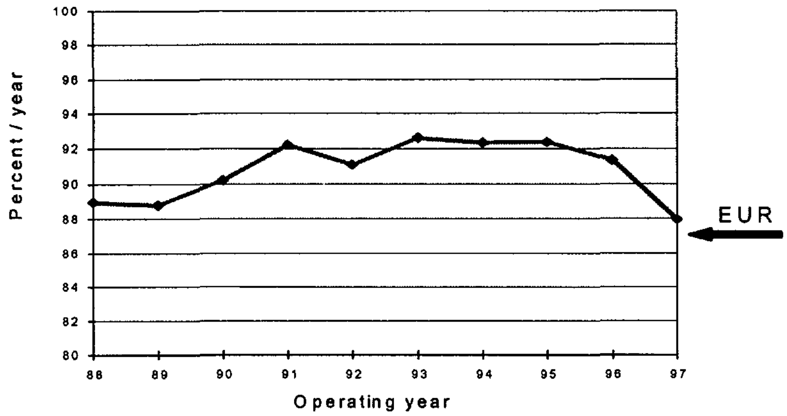

FIG. 1. Annual energy availability factors for the BWR 75 plants

The total electricity generation costs have been low, as demonstrated by the published production costs for the Forsmark 1, 2, and 3 plants during the last decade (figure 2).

3. THE BWR 90 DESIGN FOR THE 1990s

The most recent BWR design of ABB, the BWR 90, was offered commercially to Finland in 1991, as one of the contenders for the fifth nuclear power plant project in Finland.

The BWR 90 design is based on the experience from design, construction, commissioning and operation of BWR 75 plants in Finland and Sweden. Specific changes were introduced to an established reference design, that of the Forsmark 3 and Oskarshamn 3 units. Modifications were made to adapt to technological progress, new safety requirements and to achieve cost savings. An efficient feedback of operation experience, and consideration of utility requirements, was provided by a co-operation with the Finnish utility Teollisuuden Voima Oy (TVO) that operates the Olkiluoto 1 and 2 plants of BWR 75 design in Finland. These units have operated extremely well, with an average capacity factor over the last ten years of 93,2 %.

The design is characterised by the use of internal recirculation pumps, fine-motion control rod drives, and comprehensive physical separation of the four-train safety systems, basically in the same way as in its predecessor. The thermal power rating of the base version is 3,800 MWth, supplemented by a smaller unit of 3,300 MWth.

In 1997, the Steering Committee of the European Utility Requirements (EUR) group, selected the BWR 90 to be the first BWR design to be evaluated and reviewed for compliance with the EUR document.

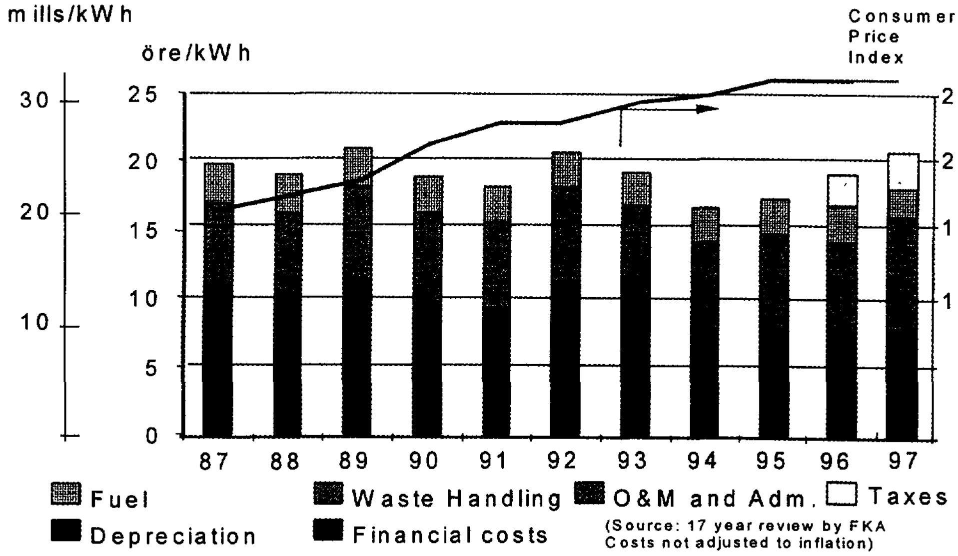

FIG. 2. Generation costs for the three BWR 75s at Forsmark

This work that is scheduled for completion by the end of 1998, has comprised a detailed assessment against the overall requirements - in Volumes 1 and 2 of the EUR document. The results will be incorporated into a "Volume 3 Subset for the BWR 90 design" document. As a matter of fact, the effort also serves to demonstrate the general applicability of the EUR document to BWR designs. The review has shown that BWR 90 meets most of the requirements; deviations mainly refer to technical details.

4. SOME BWR 90 HIGHLIGHTS

A main emphasis in the development work was to maintain "proven design" features, unless changes would yield improvements and simplifications. In line with this philosophy, the reactor design changed very little.

The reactor pressure vessel design was modified slightly; an enhanced use of large-section forgings has yielded a significant reduction in number and length of welds. This in turn reduces plant operation cost since it reduces the amount of in-service inspection to be carried out during the refuelling outage.

The recirculation system uses internal glandless pumps driven by wet asynchronous motors; this type of pump has been operating reliably in ABB BWR plants (for more than four million operating hours) since 1978. Such internal pumps have now been adopted also by other BWR vendors, in the ABWR plants.

The engineered safety systems are consistently divided into four redundant and physically separated subsystems, of which two suffice to meet the demands in any design basis accident situation. This BWR 75 concept has been reconfirmed as an optimal arrangement with respect to safety, layout and maintainability.

The safety-related electrical power supply and I&C systems are divided into four sub-divisions in the same way; the reactor protection system operates in a 2-out-of-4 logic for signal transmission and actuation.

With respect to diversity, it may be noted that the traditional ABB BWR control rod drives system incorporates diversified means of control rod actuation and insertion, by hydraulic pressure and by electrical motor. Together with a generous reactor pressure relief capacity, and combined with a capability of rapid recirculation flow rate reduction (by pump runback), it provides an efficient ATWS (Anticipated Transient Without Scram) countermeasure.

The general arrangement of the buildings *(cf. Figure 3)* is characterised by a division into a nuclear, safety-related part of the plant, containing the reactor building, the diesel buildings and the control building, and a more "conventional" part that is "separated" from the former by a wide communication area. The "conventional" part contains the turbo-generator and auxiliary systems of the plant. This arrangement is advantageous when building the plant as well as during plant operation, since the conventional part does not interfere with the nuclear part.

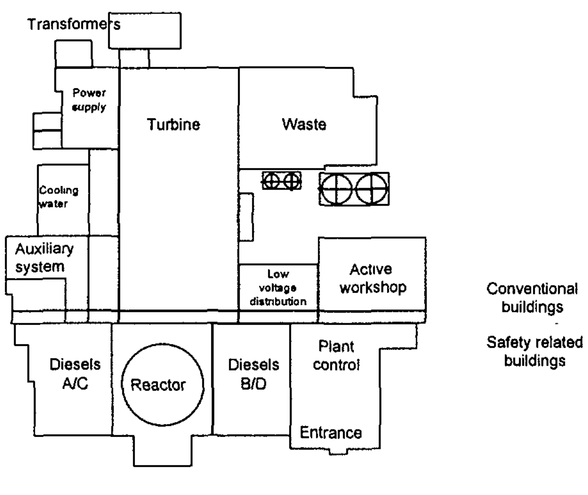

FIG. 3. BWR 90 General building Arrangement

Compared with previous plants, building volumes have been substantially reduced, yielding a significant cost reduction. Nevertheless, BWR 90, like previous plants, is characterised by a fairly spacious layout. This facilitates access to components and is a key to low occupational exposure.

The pressure-suppression containment consists of a cylindrical pre-stressed concrete structure with an embedded steel liner - as in all previous ABB BWR plants. The containment vessel, including the pressure-suppression system and other internal structural parts as well as the pools above the containment, forms a monolithic unit and is statically free from the surrounding reactor building.

At the time of the BWR 90 design, regulatory developments indicated a need to strengthen the capability of the reactor containment to withstand the effects of a core melt accident. Today, such requirements are codified in several countries, e.g., Finland and Sweden. The essential features of the BWR 90 containment to achieve enhanced environmental safety including protection during a degraded core accident are:

- The blow-down of steam to the suppression pool passes through vertical concrete pathways to horizontal openings between drywell and wetwell.
- The relief pipes from the safety/relief valves are drawn into the suppression pool via the lower drywell rather than penetrating the drywell-wetwell intermediate floor.
- A pool is provided at the bottom section of the lower drywell for the purpose of collecting and confining fuel melt debris. The pool is permanently filled with water to enhance passive safety.

In addition, the containment vessel can be vented to the stack through a filter system, installed in the reactor building, similar to the filtered venting systems installed at all nuclear power plants in Sweden. These arrangements improve the reliability of the pressure-suppression system and reduce the probability of containment leakage during a severe accident.

A simplification of the auxiliary power supply configuration is easily distinguished; the number of distribution voltage levels have been reduced. As an example, it can be noted that DC distributions at several voltage levels for power supply to control equipment has been replaced by power supply from the battery-backed AC-distribution, using distributed AC/DC converters for the supply to the various types of equipment, when needed. The simplifications of the electric power systems will of course have a significant influence on the amount of maintenance work; a substantial reduction is anticipated.

A key to modern process control and communication applied to the BWR 90 is the use of control and instrumentation systems based on micro-computers. Process communication with the control room is realised by means of distributed functional processors. These in turn interact via serial communication links with a number of object-oriented process interface units. Thus, the protection and control system configuration is characterised by decentralisation and the use of object-oriented intelligence. The arrangement satisfies the requirements of redundancy and physical separation. It includes intelligent self-monitoring of protective circuits.

The use of serial communication links guarantees interference-free performance and reduces cabling. Standardisation of the object-oriented circuits minimises maintenance and the necessary stock of spare parts. The arrangement will also tend to improve availability, since components can be replaced quickly and simply. An important aspect is that the software is also standardised to simple program functions. This makes it easy even for non-computer specialists to handle the systems, and it facilitates implementation of new micro-computer generations.

Video display units (VDUs), keyboards, and display maps are used consistently to facilitate the man-machine communication in the control room. The main control room *(cf. Figure 4)* contains several work positions, each equipped with a number of VDUs. Typically, one VDU will display a total view of the process in interest, another will provide a list of alarms, and a third VDU will display a diagram with sufficient detail to facilitate operator action. This arrangement is supplemented with a special overview panel, visible to all operators in the control room. The overview shows the main process in the form of a flow diagram and indicates the status (normal, disturbed or failed) of various plant functions by conventional instruments and computer-based displays.

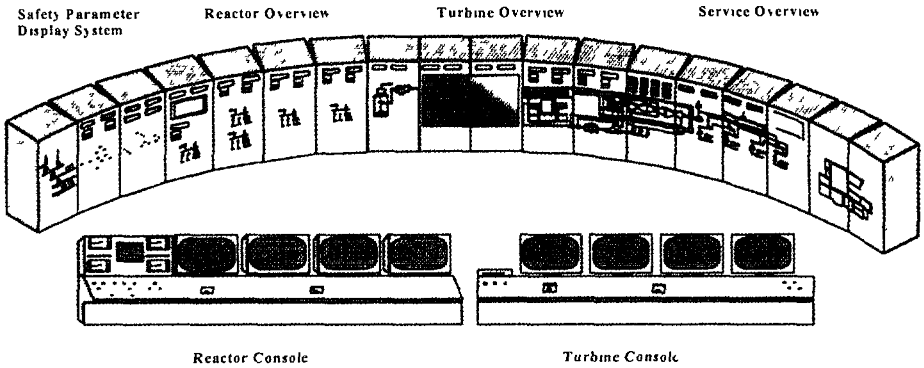

FIG 4 BWR 90 Control Room Arrangement

The main computer has the task of collecting information from the process control systems, and it communicates with the distributed micro-computers via serial links. The main computer compiles information and generates reports, such as daily/weekly operation reports, reports of periodic testing, actual status reports, and disturbance reports.

5. THE BWR 90+ DESIGN

ABB is continuing its BWR development with work on a new advanced design, the BWR 90+. The aim of the continued programme is to maintain and develop the BWR as a competitive option for a reviving market, primarily in Europe, beyond the turn of the century, as well as feeding ideas and inputs to the continuous modernisation efforts at operating plants. In essence, the programme is firmly based on evolutionary development of the company's previous, advanced BWR designs.

The main objectives of the project focus on anticipated utility needs in the 21st century; the new design should offer reliable power generation at reduced construction and operation costs and incorporate significant safety improvements. The on-going review of the BWR 90 against the EUR is of importance for the BWR 90+ development work, and observed findings are specifically addressed in the design.

The development work is conducted in co-operation with TVO, which feeds adequate operation experiences into the project and performs analyses of different design alternatives regarding safety aspects, including mitigation of severe accidents Swedish BWR operators have also recently joined the project with focus on getting ideas and inputs for modernisation and improvement works at their existing nuclear power plants.

5.1 Design and performance goals

Economic competitiveness is of paramount importance for a new nuclear power plant design. The BWR 90+ design work aims at developing a plant with: - reduced investment cost, - short construction time, - high energy availability, - short refuelling outages, - low operation and maintenance costs, - low fuel cycle cost, as well as - low waste management and decommissioning costs.

With respect to flexibility and reliability the governing design guideline again is: "Proven system design and components are to be adopted to ensure reliable electricity production, and moderate development steps are introduced only when bringing improvements." As a result, most of the fundamental design features from the previous designs with respect to the energy production capability and reliability will be incorporated also in the BWR 90+ design.

Some specific design and performance goals of the BWR 90+ development project are.

(a) Plant nominal power output; 1500 MWe
(b) Construction time; less than 1500 days
(c) Energy availability; higher than 90 %
(d) Refuelling outage; 15-20 days/year.

5.2 New evolution and safety requirements

The BWR 90+ design builds firmly on proven design, but considers and adopts new developments including new technology, digitised control equipment, and passive features and functions, as well as features that yield improved severe accident mitigation.

The design principles of BWR 90+ are based on generally established international codes and standards. In addition, specific attention is paid to the requirements of STUK, the regulatory body of Finland. The EUR documents and the EPRI URD[1] are also considered.

The safety requirements applied in the design lead to use of redundancy and diversification in addition to physical separation to ensure independence. The diversification includes use of passive systems. The design follows the "defence-in-depth" principles, aiming at ensuring:
(1) low frequency of disturbances;
(2) that disturbances are controlled and do not develop into more severe events;
(3) that accidents are mitigated and do not develop into a core melt accident; and
(4) that the effects on the surroundings of a core melt accident remain acceptable.

The most recent edition of the STUK guides[2] addresses the need for a very high safety level and calls for improved severe accident mitigation, and limited accident consequences. Some examples are:
(a) The plant shall be designed so that no release of radioactivity will occur during the first period after a severe accident, even if all easily oxidising materials in the reactor core react with water.
(b) The design should include a containment venting system, but containment venting shall not be the primary way to reduce a containment overpressure in a severe accident situation.
(c) The relief valves shall be activated only temporarily (for brief discharges) during an anticipated transient to control the reactor pressure.
(d) The plant shall be designed to prevent release of radioactive matter in case of a possible accident, including a LOCA, during shutdown conditions, e.g., resulting from human errors during refuelling.
(e) Provisions shall be made to ensure decay heat removal in the event of loss of the ultimate heat sink normally used.

In order to meet these demands, a number of changes are introduced in the BWR 90+ design, compared with previous designs, including an improved containment design, introduction of passive systems and ECCS modifications. Design measures to cope with a "degraded core" accident have been incorporated in the containment design by provision of a core catcher arrangement and filtered venting for the containment.

5.3 Improved containment design

The containment design (*cf. Figure 5*) is characterised by robust design principles. During normal operation, the containment is inerted by nitrogen gas, thereby eliminating the risk of fires during operation and the risk for hydrogen explosions in case of postulated core melt accidents.

Except for vacuum breakers, all pipe connections between drywell and wetwell have been eliminated. The number and size of the vacuum breakers have been reduced. The wetwell, including the partitioning floor, is provided with a leak-tight liner in stainless steel. This design minimises the potential for drywell - wetwell bypasses.

A dry core catcher is arranged beneath the reactor pressure vessel; its steel structure is submerged into the containment pool. In case of a severe accident, involving core melt and penetration of the reactor pressure vessel, the molten core will be collected in the core catcher, which will be cooled by the surrounding water. The containment structure is protected against the direct impacts of the molten material and does not serve as the primary barrier for a core melt. The containment proper will serve as an inherently passive system ensuring that no releases of radioactivity to the environment will occur during the first period after a severe accident

[1] EPRI URD – Utility Requirements Document of Electric Power Research Institute, USA
[2] Guide YVL 1.0 – Safety Criteria for Design of Nuclear Power Plants

with a molten core. The improved design implies a reduced risk for steam explosions, and a released molten core will be cooled in a passive way by the containment pool water.

The wetwell gas compression chamber volume and the pool water volume have been increased compared with previous designs. The improved design will accommodate the pressure build-up that may occur from hydrogen generation from all zirconium in the core for one day without activation of the cooling, overpressure protection, and support systems. Activation of active cooling systems, as well as spraying water into the drywell, will cool the containment structure, reduce the containment pressure and, in turn, prevent releases to the surrounding. The filtered venting system can be used to reduce the containment pressure in the long term without concerns for significant off-site consequences.

In the BWR 90+ design, there are no openings or pipe and cable penetrations from the lowest part of the drywell. The top of the core is located below the level of the upper drywell (or partitioning) floor. In the hypothetical case of a LOCA induced by human errors during plant shut-down and refuelling operations, the water volume in the pools above the reactor will suffice for filling the drywell volume to above the partitioning floor, and consequently, this design implies that the core will remain flooded without human action or safety system actuation.

The improved containment design – with the pools on its top - is fully adapted to construction by means of slip-forming methods; the peripheral walls of the pools are made as integral parts of the containment wall structure. Combined with an extensive use of modular building technique, this reduces the construction time and costs.

The main features of the improved containment design are:

(a) Reduced construction time and costs.
(b) Minimised probability for drywell - wetwell bypass.
(c) Core remains covered by water if loss of coolant accident occurs during refuelling.
(d) Passive core melt retention and cooling inside containment; no releases within one day in the event of a core melt accident.
(e) Containment structure is protected against core melt impact by core catcher arrangement.
(f) A dry core catcher reduces the probability for steam explosions.
(g) Core concrete interaction is negligible.
(h) Increased volumes cope with pressure build-up from hydrogen generation at core melt accident.
(i) Nitrogen gas inertion allows cooling and pressure reduction by water spraying without risk for hydrogen explosions.
(j) Ultimate overpressure protection by filtered containment venting.

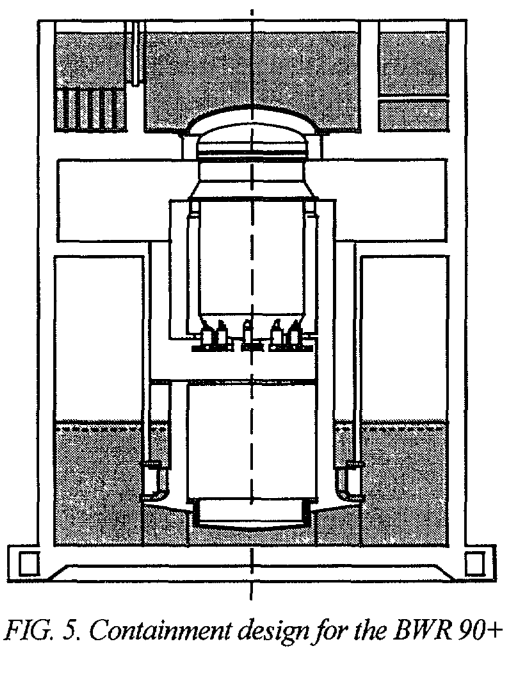

FIG. 5. Containment design for the BWR 90+

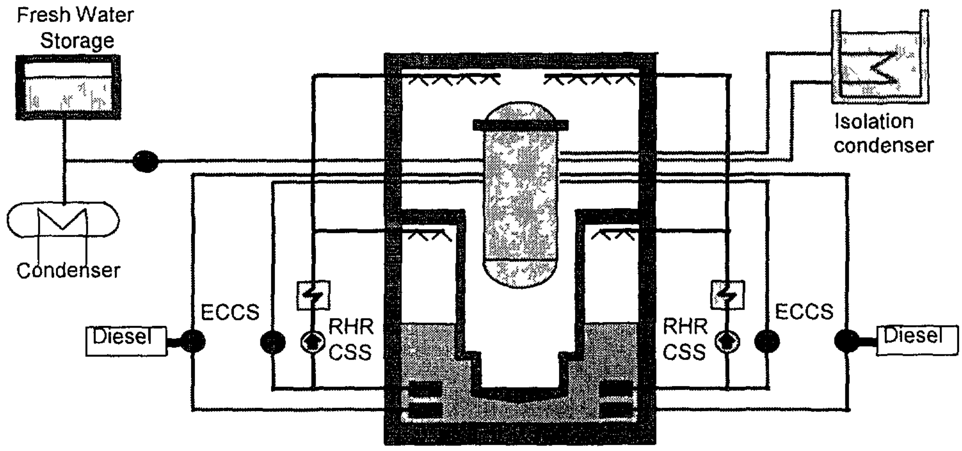

FIG. 6. Decay heat removal and coolant make-up systems.

5.4 Decay heat removal and safety systems

A number of diversified components and system functions have been introduced, in line with the new safety requirements from the Finnish safety authority STUK, which demand diversity and provisions, in particular against loss of the final heat sink normally used, and that relief valves be actuated only temporarily. To this end, a passive heat removal system similar to the isolation condenser in the first ABB Atom NPP, Oskarshamn 1, will be incorporated in the design. The principles for the decay heat removal and coolant makeup systems are outlined in figure 6.

6. CONCLUSIONS

ABB is convinced that there will be a reviving market for new nuclear power capacity in the near future. ABB is determined to be in a position to be able to compete for the new orders and therefore has continued its development efforts.

The BWR 90 design is available today for deployment in new plant projects and has been offered commercially. The design has, with a positive outcome, been subjected to a review by European utilities to evaluate how it compares with the requirements established by the EUR group.

The development of the BWR 90+ design is based on its 'forerunners' and can be referred to as a 'proven design', in line with power utilities' preferences. The 1500 MWe design incorporates considerations of new safety requirements, including severe accident impacts, and it will be marketed by the turn of the century. The development work on the BWR 90+ design will also serve as input for modernisation, uprating and improvements of earlier generations of nuclear power plants.

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THE DEVELOPMENT OF ABWR

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Abstract

The first Advanced Boiling Water Reactor (ABWR) started commercial operation as Tokyo Electric Power Company's (TEPCO) Kashiwazaki-Kariwa Nuclear Power Station Unit No.6 (K-6) in November 1996 and its sister Unit No.7 (K-7) in July 1997. The ABWR was developed to achieve higher reliability and safety margin while improving overall operability and economics. To achieve these goals, the optimal Boiling Water Reactor (BWR) technologies had been studied, tested and were finally adopted into the ABWR design. These technologies were called "First of a Kind" and include the Reactor Internal Pump (RIP), Fine Motion Control Rod Drive (FMCRD), Reinforced Concrete Containment Vessel (RCCV), and integrated digital Instrumentation and Control System (I&C). Intensive development study, confirmation tests and verification tests were conducted by the plant equipment suppliers, electric utilities, and government agencies. During plant construction, the system and equipment functions and operational characteristics were confirmed through various tests. Before the start of the commercial operation, it was confirmed that these first ABWR units met all the design goals that had been established for the next generation of nuclear power plants. Both units have now completed the first fuel cycle of operation without any unplanned plant shut down and have achieved very high availability factors. This paper describes the development and construction of these first ABWR units in the world.

1. INTRODUCTION

The first Advanced Boiling Water Reactor (ABWR) started its commercial operation when Tokyo Electric Power Company's (TEPCO) Kashiwazaki-Kariwa Nuclear Power Station Unit No.6 (K-6) was taken into operation in November 1996; its sister Unit No.7 (K-7) followed in July 1997. An outlook of the two plants is shown in figure 1.

1.1 Development Purpose

Based on the long construction and operation experience that has been accumulated in the USA, European countries, and Japan, the Japanese BWR utilities and BWR equipment suppliers identified the need to develop the next generation of BWR as a means of achieving broad scope improvements throughout the nuclear power plant design, construction and operation. High level goals were established for this development effort and they were: higher levels of safety and performance, improved operability, and cost effectiveness. To achieve these goals, the following detailed objectives were established:

- Core Damage Frequency less than 10^{-7}
- Construction period from Rock Inspection to Commercial Operation less than 48 months
- First Refueling Outage period less than 55 days
- Occupational Radiation Exposure less than 0.36 man-Sv per year

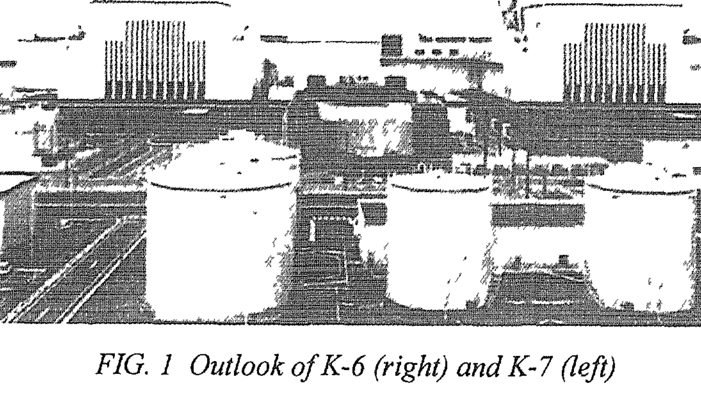

FIG. 1 Outlook of K-6 (right) and K-7 (left)

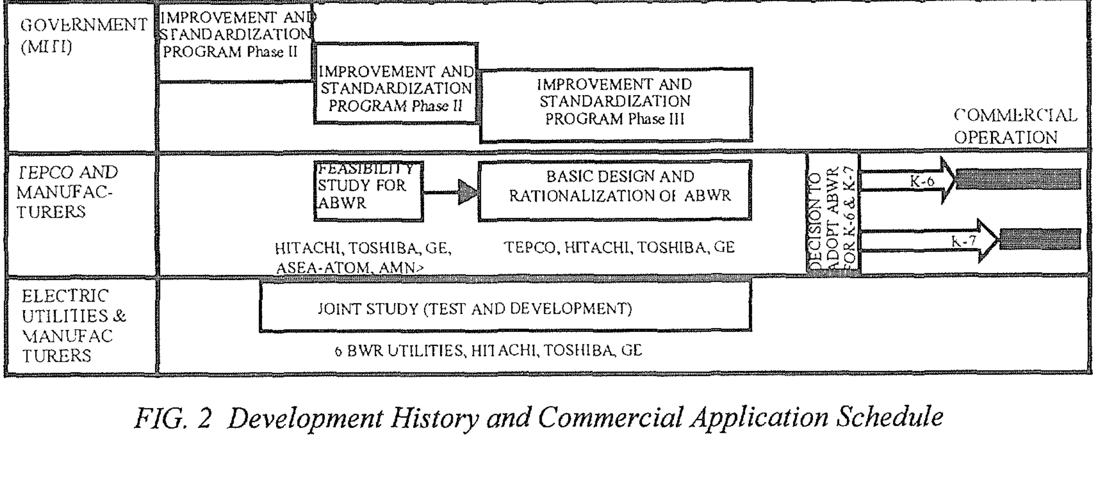

FIG. 2 Development History and Commercial Application Schedule

1.2 Development History

About 20 years ago, TEPCO saw a need to combine the best BWR technologies from around the world in order to aggressively pursue the development of the most reliable, maintainable, and efficient BWR possible. To implement TEPCO's idea, the world's BWR suppliers formed an Advanced Engineering Team (AET) consisting of GE of USA, Hitachi and Toshiba of Japan, ABB Atom (formerly ASEA-ATOM) of Sweden, and Ansaldo MN from Italy. The feasibility of this ABWR project was studied by the AET, and the result was presented to TEPCO. Upon receipt of this report, TEPCO decided to proceed with the development of the ABWR. A series of ABWR joint study programs, between the six (6) Japanese BWR utilities and GE, Hitachi, and Toshiba BWR suppliers, were conducted in which key technologies were studied and verified. At about that time, the ABWR was adopted in the Japanese government's Third Improvement and Standardization Program for nuclear power plants. Through these development programs, the ABWR feasibility was fully demonstrated, and TEPCO announced the adoption of ABWR for their K-6 and K-7 units in 1987. The development history is shown in figure 2.

1.3 Key Technologies

Many new technologies called "First of a Kind" were scrutinized, tested, and demonstrated. Finally, the following systems and equipments were selected for application to the ABWR.

The Reactor Internal Pump (RIP) was adopted for the reactor recirculation system in place of the conventional external loop system. The external loop piping and large pumps are replaced with ten (10) RIPs, which are directly attached to the bottom of the Reactor Pressure Vessel (RPV).

The Fine Motion Control Rod Drive (FMCRD) was developed and adopted in place of the conventional Locking Piston Control Rod Drive (LPCRD). The FMCRD is driven by a stepping motor for the normal operation, and a hydraulic system is utilized for the emergency scram operation.

The reactor Primary Containment Vessel (PCV) has typically been a thick steel containment vessel in past designs but, for the ABWR the Reinforced Concrete Containment Vessel (RCCV) was adopted. In this design, a thin steel liner plate provides the leakage prevention function while the 2-meter thick reinforced concrete provides the pressure containment function.

Three complete divisions of Emergency Core Cooling Systems (ECCS) are provided for a total compliment of three high pressure core injection systems and three low pressure core injection systems.

Large low pressure turbine with 52-inch last stage blades and Moisture Separator ReHeater (MSH) were adopted for improved turbine system thermal efficiency. Two types of heater drain forward pumping systems, i.e., the High Pressure Drain Pump system (HPDP) and the Low Pressure Drain Pump system (LPDP), are used to achieve better thermal efficiency.

Integrated digital control and instrumentation system architecture was adopted for the first time. Control and instrumentation signals are converted into digital signals, and they are conveyed through series of fiber optic multiplex data links. The main control room panel is totally redesigned from a human factors viewpoint. Plant operations can be achieved at the main control console by touch operation on Cathode Ray Tubes (CRT) or flat-panel displays. Major plant parameters are shown on the wide display panel so that all the operators in the control room can share the most important information at the same time. All the annunciators are displayed by different colors according to its importance.

Out of these key ABWR technology developments, the RIP, RCCV, and Instrumentation and Control are described in further detail in this paper.

2. REACTOR INTERNAL PUMP

2.1 Development Objective

A conventional BWR uses two (2) external pumps and ten (10) sets of jet pumps for the reactor cooling water recirculation purpose called the Primary Loop Recirculation (PLR) system. However this PLR system required a large primary containment volume to house those external pumps and piping. Also, that external loop piping can be the source of high occupational radiation exposure for the maintenance crew. In consideration of these points, direct mounting of the recirculation pumps to the RPV bottom head was considered.

2.2 Joint Studies

The RIP technology was developed in Europe, and has been used in the European BWRs for years. However, due to the differences in regulations and/or industrial practice in Japan, it was necessary to test the RIPs to study their applicability in Japanese BWRs. Two sets of RIP were purchased from a European vendor and they were tested in Japanese test stands as part of a series of utility-supplier joint study program. Included within those programs were: conceptual design of RIP system, pump verification tests, seismic evaluations, handling equipment development, and core flow related studies. At the same time, the core transient and accident operations with RIP were tested and analyzed. The test results were incorporated into the core transient simulation computer program.

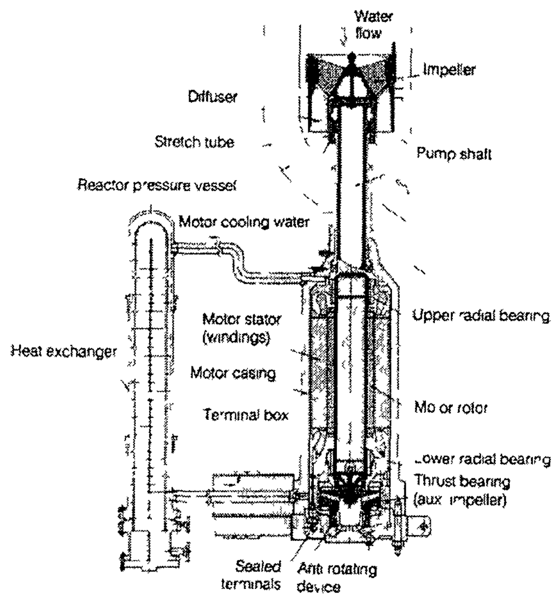

FIG. 3 Reactor Internal Pump (RIP) cross-section

2.3 Government Tests

The government of Japan, Ministry of International Trade and Industry (MITI), has conducted the Improvement and Standardization Program (ISP) for the improvement of the Light Water Reactor (LWR) based on the construction and operation experience in Japan. The third phase of that ISP focused on the development of the advanced BWR and PWR.

As a part of the third ISP, RIP verification testing was conducted at the Nuclear Power Engineering Corporation (NUPEC) in Japan. Two imported RIPs were tested at the NUPEC test loop as a performance Verification Test. The test loop had a full scale 60 degree sector of the ABWR vessel with full size lower plenum reactor internal equipment mocked-up. [An overview of the RIP arrangement at the bottom of the RPV is shown in figure 3.] Through that series of tests, the applicability of the RIP system to the ABWR was confirmed by the government agency. The test results were issued as a public document.

During the Verification Tests, Japanese suppliers Hitachi and Toshiba started the development of the RIP applying their own technology. When these Japanese RIP developments were completed, they were used for the NUPEC Proving Test. This Proving Test was conducted to confirm the integrity of the welds that join the RIP casing and the RPV to withstand the effects of pump induced vibrations; the integrity of the RPV nozzle to which the RIP casing is welded to withstand the effects of thermal and mechanical stress associated with the increased flow; and the integrity of the welds that join the Reactor Internals and the RPV to the effects of Flow Induced Vibration (FIV) under anticipated increased core flow operating condition.

2.4 Japanese Supplier Development

Japanese suppliers Hitachi and Toshiba developed their own RIP designs based upon extensive component testing. Many types of thrust and axial bearing designs were tested, and the bearings which had the best stability characteristics were selected. To achieve high efficiency and optimal head/flow characteristics, many hydraulic models of the Impeller and Diffuser were developed. To minimize the leakage potential from the RPV, the entire RIP (including motor) are enclosed within the RPV primary pressure boundary. Motor winding integrity in the high temperature and high pressure water condition was confirmed through tests. An anti rotation device was developed to avoid RIP reverse rotation when one pump stops while other pumps continue operation.

When all component design was complete, a full size prototype RIP was assembled and tested at the supplies' in-house test stand. A series of tests under both normal operating conditions and abnormal plant conditions were conducted which, confirmed that the RIP that had been developed had adequate performance and characteristics suitable for the actual plant application. After these in-house tests, these RIPs were used for the Proving Test conducted at NUPEC.

At the same time, 1/5 scale model tests were conducted. Tests were done with one (1) pump to confirm single pump performance characteristics, and ten (10) pump tests were conducted to demonstrate the in vessel flow dynamics and expected stress levels due to Flow Induced Vibration. Also, interactive effects, such as pump re-start under reverse flow conditions was confirmed.

Prior to shipment of the actual production units, all RIPs were tested at the in-house test stand. They were tested with a power supply system exactly like that which is used in the actual plant installation. Each RIP is connected to one Adjustable Speed Drive (ASD), and three (3) ASDs are connected to one Motor-Generator (M-G) set. RIP characteristics, such as head/flow curve or pump efficiency, were confirmed. Since the in-house test stand can test only one pump at a time, RIP tests that address the interactive affects between pumps in operation can only be done after the site installation. Partial pump operation condition and simulated accident conditions were tested before the fuel loading. After the fuel loading, reactor power was raised progressively to 20%, 50%, 75%, and 100% power condition. At each power condition, tests at anticipated plant transient conditions were conducted, and the test results were compared with the analysis results or the model tests that had been previously conducted.

Through the completion of these tests, it was confirmed that the newly developed RIPs can achieve expected performance before the start of the commercial operation.

3. REINFORCED CONCRETE CONTAINMENT VESSEL

3.1 Development Objective

Conventional BWRs use 38mm thick steel plates to construct the Primary Containment Vessel (PCV). The PCV is designed to prevent the release of non-condensable gases and steam postulated to occur under the conditions of a design basis Loss of Coolant Accident (LOCA). As a steel containment, there are some restrictions on how such a vessel can be configured. Also, during the construction, building concrete can only be poured after the leak test of the steel PCV has been confirmed. To improve these condition, the adoption of the RCCV was considered. The RCCV has been previously applied in some BWR-6 plants outside of Japan.

With the application of the RCCV, the accident condition pressure is sustained by two 2-meter thick reinforced concrete, while gas leakage is controlled by a 6.4 mm thick steel liner plate.

3.2 Joint Studies

The series of Joint Study titled "Evaluation on RCCV Configuration and Confirmatory Test to Establish Code" was conducted among the BWR utilities and plant suppliers, with civil contractors providing technical support. Extensive tests were conducted on the each elements of the RCCV including full sector modeling. Test items are listed below.

- Structural integrity tests
  - Shear stress test on reinforced concrete
  - Test on the RCCV openings
  - Test on the RCCV top slab
  - Test on the RCCV brackets and its anchor
  - Test on the structural characteristics on RPV pedestal
  - Test on the rebar joints
  - Full sector 1/6 scale model

- Liner and Penetration tests
  - Shear stress test on liner anchor
  - Thermal impact test on liner and liner anchor
  - Thermal load test on high temperature piping penetration
  - Stress test on penetration anchor

With the results of these tests, the design base for the RCCV were established and the basic data, to be used for the establishment of regulatory licensing guidelines by the Japanese government, was developed.

3.3 Site Installation and Testing

The RCCV steel liner plate is pre-assembled in the shop in sections as large as possible, within the size of transportation restrictions. The sections were pre-assembled into twenty nine (29) meter diameter cylindrical units at a location immediately adjacent to the reactor building and directly carried into the building using a 925 ton capacity large mobile crane. The concrete is poured around the RCCV liner plate after the rebar is installed. The reinforced concrete of the RCCV is integrally interconnected with the concrete floors and walls of the Reactor Building. Therefore, the construction of the RCCV can be done simultaneously with the building construction.

The Structural Integrity Test (SIT) is a test to confirm the extent of swelling expected when the RCCV is pressurized and also to confirm that crack propagation under such condition is within acceptable limits. The SIT was not required for the conventional PCV because of the configuration difference. Full scope of SIT measurement was done only for the first unit (K-6). For subsequent unit (K-7), SIT without strain measurement was applied. The measurements at K-6 showed that the actual swelling of the inner diameter was not as predicted due to conservatism in those calculations.

The Air Pressure Test and Leak Rate Test are conducted to confirm the vessel integrity and leak rate under pressurized condition. These tests have been done in the past on conventional BWRs but the test timing is much earlier for the RCCV. Actual test results demonstrated that the leak rate was well within the specification limit.

4. INTEGRATED DIGITAL INSTRUMENTATION AND CONTROL

4.1 Development Objective

To meets strong needs for higher reliability and cost-performance, the advanced control and monitoring system has been developed by rationalizing the conventional technology in the field. The rationalization was done through the utilization of reliable digital technology and optical information transmission technology, and others, which are now commonly used in computer applications. The goal of the development work is to ensure safe, stable operation of the plant and to secure harmony between man and machine.

The latest electronic devices are employed to create an advanced human interface to alleviate the burdens of the operators, and digital technology has been applied to the safety related systems based upon sufficient operational experience of non-safety digital systems in order to further enhance reliability and maintainability of the plant.

4.2 Joint Study

Several joint studies between Japanese BWR utilities and Japanese suppliers were conducted. These studies focused on operational improvement using man-machine interface technology, control and information system configuration, application of digital technology into the safety related systems.

At the same time, plant operator work load was analyzed to determine the best main control room configuration. Series of interviews to the operators and motion analysis were conducted. The

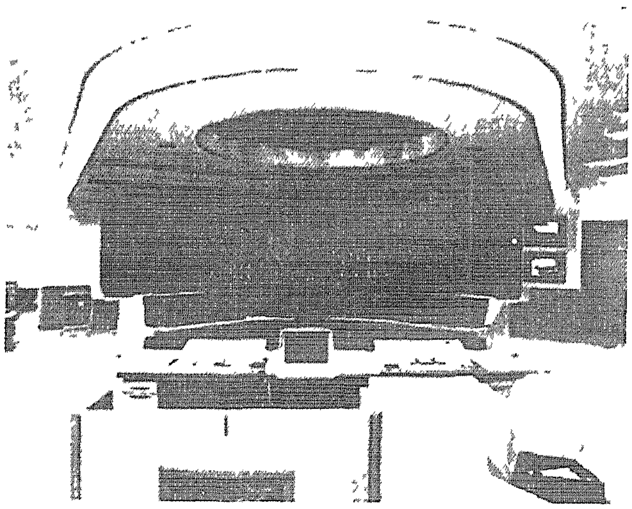

FIG. 4 K-7 main control room panels

results of these studies were combined into the partial mock up test panels, with which more detailed analysis was conducted. Finally full size wooden mock up test panel was made for the operability confirmation by the operator.

4.3 Human Interface

The design philosophy has two main points: 1) to further facilitate and ensure monitoring and operation, and 2) to make plant operation more efficient. To attain these goals, the control board configuration consists of a main control console and a wide display panel (cf. figure 4). The display panel presents, simultaneously, important information on the plant to all of the operators involved, while the compact main control console provides integrated information for normal monitoring and operation.

These consoles and panels have flat-panel displays as well as CRT displays with touch operational support to realize compact human interface. The alarm system is another feature. Conventionally, there are about 1600 windows provided, whereas on the display panel about 70 essential alarm windows and about 140 system alarm windows selected from among many that are arranged to furnish collective information of the plant. When a particular hardware window begins to blink, the CRT and flat-panel displays present detailed information in a hierarchy. In order to facilitate the operators' understanding of the plant status, each alarm window is designed to display in three colors the degree of various effects on the operation.

With the adoption of these new features, it was very important to train plant operators before the ABWR plant went into the pre-operational stage. The full size plant simulator with various condition simulation function was installed in BWR Operator Training Center Corporation which is jointly owned by the Japanese BWR electric utilities and the BWR suppliers. TEPCO operating crew have been well trained in the facility before the actual plant operation.

4.4 Verification and Validation

From the Quality Assurance / Quality Control view point, it is very important to confirm the adequacy of the software. For this purpose, Verification and Validation (V&V) is conducted for the safety-related systems. V&V is defined in Japan in technical standard JEAG-4609 (Application Criteria for Programmable Digital Computer System in Safety-Related Systems of Nuclear Power Plants) of the Japan Electric Association. Besides utilization of POL(Problem Oriented Language) and simple software structure have made it easy to understand the function and review its quality.

V&V is conducted step by step as shown below :

- Verification of System Design Specification
- Verification of Software Design Requirement Specification
- Verification of software design and implementation
- Verification of integrated hardware and software
- Validation tests

5. OTHER FIRST OF A KIND TECHNOLOGY

As mentioned in Article 1.3, Key Technology, there are several other technologies that have been applied for the first time in Japan with the ABWR. A brief explanation of other key technologies is provided below.

5.1 Fine Motion Control Rod Drive

BWR plants apply bottom entry Control Rods (CR). Conventional BWRs utilize hydraulically operated Locking Piston Control Rod Drive (LPCRD) and these LPCRDs position the CR within 152.4 mm increments.

To reduce the thermal impact on the fuel associated with the insertion and the withdrawal of the CR, a finer CR positioning was desired. The FMCRD applies a ball nut and screw which is driven by a stepping motor, and has the ability to position the CR at increments of 18.3 mm. Normal CR withdrawal and insertion is done by the stepping motor rotation while emergency CR insertion, or scram, is accomplished by separate hydraulic power.

With the application of the electrical motor, simultaneous CR withdrawal/insertion became possible. This new features enables a significant reduction of the plant start-up time. The capability to insert the CR with the electric motor provides a redundant means for the assuring CR insertion.

5.2 Three Division Emergency Core Cooling System

Conventional BWRs typically have one (1) High Pressure Core Spray system, one (1) Low Pressure Core Spray system, and three (3) Low Pressure Core Injection system for the Emergency Core Cooling System (ECCS). To enhance the high pressure core injection capability and its redundancy, three (3) fully separated high and low pressure core injection systems were incorporated in the ABWR. Two (2) High Pressure Core Flooder systems, and one (1) Reactor Core Isolation Cooling system are used for high pressure core injection. Three (3) divisions of Low Pressure Flooder system as well as for residual heat removal at plant shutdown.

Each ECCS division is connected independently to a safety-grade emergency diesel generator, in addition to the normal electric power bus. Each ECCS, of its own, has 100% capacity of the core injection. Thus, emergency core injection for high- and low-pressure condition is highly assured.

5.3 High Efficiency Turbine System

The 1,100 MW class turbine generator was the maximum capacity utilized in nuclear power plants in Japan. To improve the thermal efficiency, the large 1,350 MW class turbine-generator, which has fifty-two (52) inch long last stage low pressure turbine blades, was first introduced. This fifty-two (52) inch blade was the largest blade of this kind and so, basic tests on high/low speed dynamic balance test and vibration tests were conducted.

The Moisture Separator Reheater (MSR) was adopted instead of the conventional Moisture Separators (MS). Two (2) stage heating by High Pressure (HP) turbine extract steam and Main Steam extract steam heats up the Main Steam before entering the Low Pressure (LP) turbine. Full scale mock-up testing was conducted as a utility-supplier joint study. Thermal and mechanical characteristics data were acquired through these tests.

Condensed water is heated up by four stages of low pressure feed water heaters and two stages of high pressure feed water heaters. Turbine extraction steam or Main Steam extraction steam is used to warm the condensed water. Condensed heating steam is drained to the heater drain pump system. The High Pressure Drain Pump system forwards the condensed water into the suction side of the Reactor Feed water Pump. The Low Pressure Drain Pump up system forwards the condensed water into the suction side of the Condensate Demineralizer.

6. QUALITY ASSURANCE

It is well known that a new design may be vulnerable to unforeseen problems. Since the ABWR adopted totally new systems and technologies in many areas, a careful and systematic development and verification program was applied to preclude such problems.

6.1 Test Before Use

Throughout the ABWR development, the attitude of "Test Before Use" was the key word for all the new technology. Imported technology, even though it may have had significant enough operation experience outside of Japan, went through series of tests by the ABWR suppliers. Through these tests, important knowledge and experience was accumulated by the ABWR suppliers.

6.2 Design Review

In addition to each plant supplier's internal design review, TEPCO supervised design reviews (called "Juuten Sekkei Review") were conducted. About thirty (30) systems and major pieces of equipment were selected for the Juuten Sekkei Review. During these reviews, all the pointed failures or malfunctions were identified, systematically recorded and then re-confirmed either by analysis, shop unit test, shop combination test, pre-operational test, or startup test. During the plant pre-operational or startup testing, the completion of these Juuten Sekkei Review derived activities was one of the check points that must be cleared before going onto the next phase of the test program.

7. CONCLUSION

The ABWR was developed based on the BWR technology that has evolved in many countries. However the success of the development depended significantly on the long years of confirmation and verification tests conducted jointly by the Japanese government, Japanese utilities, and BWR suppliers.

The world's first two ABWRs have started commercial operation, they have completed their first full fuel cycle of operation without any unexpected plant shut downs, and the first refueling outages were successfully completed. The ABWR development goals have been achieved by these first units. With these demonstrated superior plant characteristics, two more ABWRs are now under licensing review in Japan, and many more ABWRs are planned.

Outside Japan, the ABWR was awarded the design certificate from the U.S. Nuclear Regulatory Commission, and the construction of two ABWRs plants has begun in Taiwan, China. The ABWR has opened the door for the future of nuclear power generation, and the new wind will circulate all over the world.

SWR 1000: THE NEW BOILING WATER REACTOR POWER PLANT CONCEPT

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Abstract

Siemens' Power Generation Group (KWU) is currently developing - on behalf of and in close co-operation with the German nuclear utilities and with support from various European partners - the boiling water reactor SWR 1000. This advanced design concept marks a new era in the successful tradition of boiling water reactor technology in Germany and is aimed, with an electric output of 1000 MW, at assuring competitive power generating costs compared to large-capacity nuclear power plants as well as coal-fired stations, while at the same time meeting the highest of safety standards, including control of a core melt accident. This objective is met by replacing active safety systems with passive safety equipment of diverse design for accident detection and control and by simplifying systems needed for normal plant operation on the basis of past operating experience. A short construction period, flexible fuel cycle lengths of between 12 and 24 months and a high fuel discharge burnup all contribute towards meeting this goal. The design concept fulfils international nuclear regulatory requirements and will reach commercial maturity by the year 2000.

1. INTRODUCTION

Almost right from the start, the development of boiling water reactors (BWRs) in Germany has been characterized by major innovations. In 1968, for example, the world's first fine motion control rod drive was installed at Lingen. The Brunsbüttel reactor was then the first in the world to be equipped with internal reactor water recirculation pumps, and the twin-unit plant Gundremmingen B and C, which started commercial operation 14 years ago, incorporated all main features of a so-called advanced BWR; namely: fine motion control rod drives, internal recirculation pumps, a three-train full-range residual heat removal (RHR) system and a cylindrical pre-stressed concrete containment with steel liner. These innovations set examples for BWR development efforts world wide.

Due to the good availability ratings achieved by BWR plants, further development of the BWR product line was resumed in Germany as well in 1992. Since then, Siemens - under an order from and in close collaboration with the German nuclear utilities and, since 1995, with support from European partners (TVO of Finland, KEMA of the Netherlands, PSI of Switzerland and ENEL of Italy) - has been designing an advanced BWR plant with an electric output of around 1000 MW.

What was the motivation behind further development of the BWR product line?

There were two principal reasons for this: On the one hand, the level of plant safety should be raised even further; i.e. the probability of occurrence of a core melt accident should again be considerably reduced. Also, in accordance with requirements stipulated in the German Atomic Energy Act, a postulated core melt accident should be controlled in such a way that there would be no need for emergency response actions in the vicinity of the plant. On the other hand, nuclear power generating costs should be improved, compared to those of other energy sources, as they had recently been suffering a marked disadvantage due to the drop in fossil fuel prices and because of increasingly stringent safety requirements and higher maintenance expenses. Hence, there were two objectives which appear, at first sight, to be quite paradoxical; namely:

- To increase plant safety, and
- To reduce plant construction and operating costs.

These objectives can only be met through appropriate simplification - both in plant safety systems and in the systems needed for normal plant operation.

I shall now be describing the main design features of the SWR 1000 that have enabled these objectives to be met.

2. SAFETY CONCEPT WITH PASSIVE SAFETY EQUIPMENT

The safety systems installed until now in operating nuclear power plants have consisted of systems that are controlled by a complex reactor protection system and that need an uninterruptible supply of electric power.

It is not possible to increase plant safety any further by designing these safety systems with even more multiple redundancy than they have had in the past - in fact, such an approach would be economic nonsense. The solution instead is to introduce passive safety systems which can serve as substitutes for some of the existing active systems and which do not need any instrumentation and control (I&C) equipment or an external power supply to operate, but which function instead by means of basic laws of nature such as gravity, heat transfer and natural convection (Figure 1).

A definition of passive systems is given in IAEA-TECDOC-626. How this approach has been applied to the design concept of the SWR 1000 is shown in Figure 2. Here you can see that the SWR 1000 uses a variety of different kinds of passive components. These include, for example, the emergency condensers provided for passive heat removal from the reactor pressure vessel (RPV), and the containment cooling condensers for passive heat removal from the containment - both types of components being designed to function without any activation signals and without any moving parts (Category B). The next category (Category C) contains the passive flooding lines with their check valves, the safety-relief valves with spring loaded pilot valves and feedwater isolation valves. The components covered by Category D are allowed to be activated by external means, although in the design concept for the SWR 1000 this can comprise either activation by safety I&C equipment using

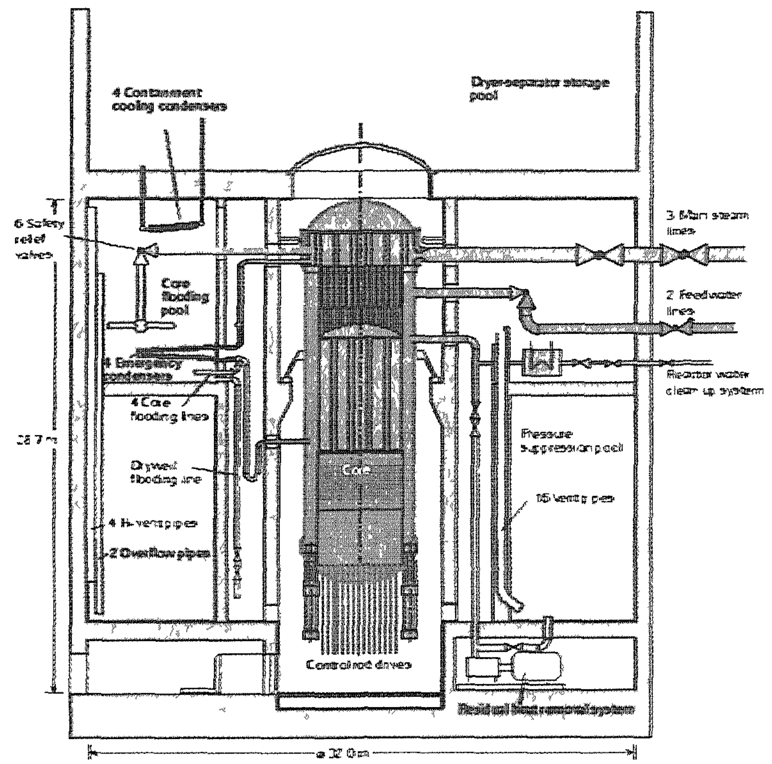

FIG. 1 Containment

Category	A	B	C	D
Examples	• RPV • Piping • Vessels	• Emergency condenser • Containment cooling condenser • Passive pressure pulse transmitter	• Check valve • SRV • Spring-loaded pilot valve	• Scram system
Structure (pressure retaining)	X	X	X	X
Moving working fluid		X	X	X
Moving mechanical parts			X	X
Stored energy • Battery • Compressed fluid • Elevated fluid			X	X
External signal				X

FIG. 2 Categories of Passive Features

solenoid valves, or activation by the passive pressure pulse transmitter system using diaphragm valves, the latter system being completely independent of plant I&C equipment. Category D contains the scram system, the safety-relief valves and the main steam isolation valves with solenoid pilot valves.

Figures 3 and 4 show which systems are used to perform the required safety functions.

By combining well-known active safety equipment with passive safety systems of diverse design, the effects of Common Cause Failures are significantly reduced and the frequency of core damage states caused by plant-internal events is about two orders of magnitude lower than that of contemporary plants. In fact, the integral frequency of core damage states calculated by proven methods for initiating events occurring during power operation is only 5.2×10^{-8} per year.

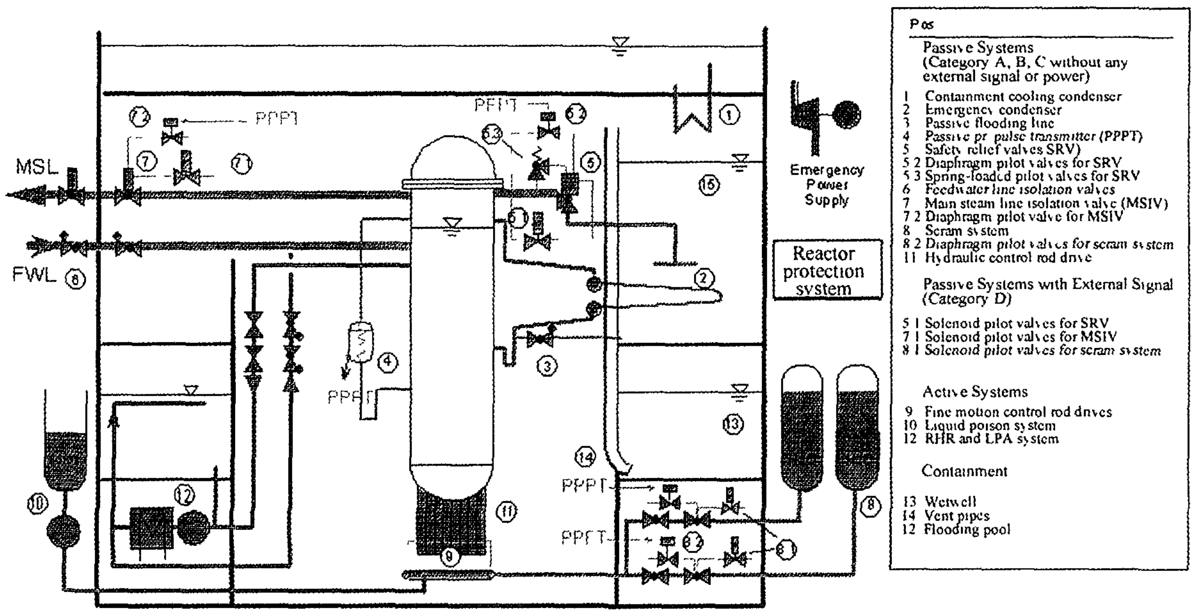

FIG. 3 Active & Passive Systems

Safety functions		Passive systems (classification acc. IAEA TECDOC626)		Active systems
Reactivity Control		Scram system 4 tanks (H_2O) Initiations: - Diaphragm pilot valve, PPPT (Passive C) - Solenoid pilot valves (Passive D)		137 Fine motion control rod drives Liquid poison system
Containment isolation MSL		2 MSIV per steam line Initiations: - Diaphragm pilot valve, PPPT (Passive C) - Solenoid pilot valves (Passive D)		1 Gate valve (if necessary)
Reactor pressure control		6 Safety relief valves Initiation: -Spring loaded pilot valve (Passive C) -Solenoid pilot valve (Passive D)		
Reactor depressurization		6 Safety relief valves -Diaphragm pilot valve/PPPT (Passive C) - Solenoid pilot valves (Passive D)	4 Emergency condenser (Passive B)	
RHR from RPV	HP LP	4 Emergency condensers (Passive B)		
Core flooding	LP	4 Flooding lines (Passive C)		2 RHR and LPCI systems
RHR from containment		4 Containment condensers (Passive B)		

PPPT - Passive pressure pulse transmitter

FIG. 4 Passive and Active Systems for Accident Control

From a deterministic analysis point of view, taking single failures into account, this design concept also allows all postulated design basis accidents to be controlled with the passive systems alone. This has been analytically verified using the validated computer code RELAP 5, (Figure 5). The analyses showed that in the event of any accidents, even those involving loss of coolant, core cooling can always be maintained and at no time will there be any increase in the fuel rod cladding temperature. By way of example, Figure 6 shows the results of an analysis based on a feedwater line break controlled exclusively using passive systems.

Analysed Accidents	Effective Systems	Results
Transients: • Loss of - main heat sink - auxiliary power supply - *feedwater*	• passive scram activation by passive pressure pulse transmitters • passive reactor pressure relief (spring-loaded pilot valves) • passive residual heat removal via emergency condenser	• no increase in fuel temperature • RPV-pressure remains • < 1.1 x design pressure
ATWS:	• passive activation (spring loaded pilot valves) of 5 out of 6 safety-relief valves • control rod insertion by electric drives within 90 seconds	• no increase in fuel temperature • RPV-pressure remains < 1.3 x design pressure
LOCA: • Main steam line break (1050 cm²) • Feedwater line break (707 cm²) • Emergency condenser line break (177 cm² and 11 cm²) • *Rupture of RPV lower head* (15 cm²)	• passive scram activation • passive activation of Main steam line isolation and depressurization *(5 out of 6 safety-relief valves)* • emergency condensers (3 out of 4 in the event of a break) • passive core flooding	• no increase in fuel temperature • maximum containment pressure is 2,6 bar

FIG. 5 Results of Transient and LOCA Analyses

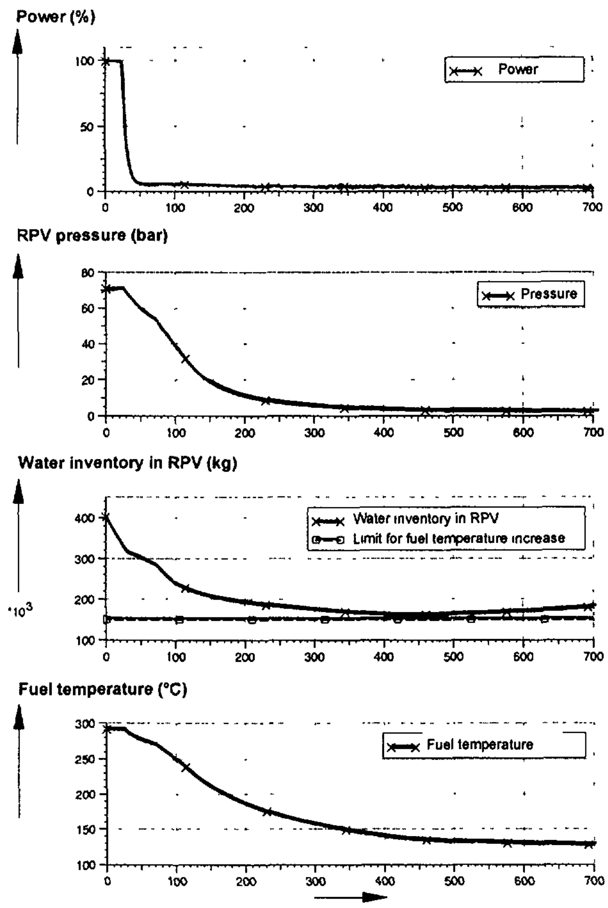

FIG. 6 LOCA Control with Passive Systems only

The main prerequisite for ensuring effective operation of the passive systems is the containment design which is specially adapted to these systems. This design incorporates flooding pools situated above the pressure suppression chamber which serve not only as a heat sink for the emergency condensers but also as a water reservoir for passive flooding of the reactor as well as flooding of the drywell for cooling the exterior of the RPV in the event of a core melt accident.

The dryer-separator storage pool, which is full of water at all times, serves as a heat sink for removing heat from the containment. The water inventory stored in this pool is sufficient for around 5 days of passive heat removal. Only after this period of time has elapsed would the water inventory have been completely evaporated, requiring a supply of makeup water provided by means of manual actions, e.g. using fire brigade pumps. During this space of time, passive core cooling and heat removal are possible without any need for I&C equipment or a power supply. Thus, the safety systems of the SWR 1000 provide for entirely passive post-accident core cooling and heat removal over a period of several days, and by extremely simple means at that.

Compared to traditional, active residual heat removal (RHR) systems comprising an interconnected chain of RHR, component cooling and service water systems, I&C equipment and power supplies (including emergency diesels), the passive equipment such as the emergency condensers and the containment cooling condensers - which are basically just heat exchangers with inlet and outlet lines - are much cheaper, not only in terms of construction cost but also in terms of their operating costs which consist of expenses for inspection and maintenance work. Because of the large water inventory contained inside the RPV itself, there is also no longer any need for high-pressure coolant injection systems.

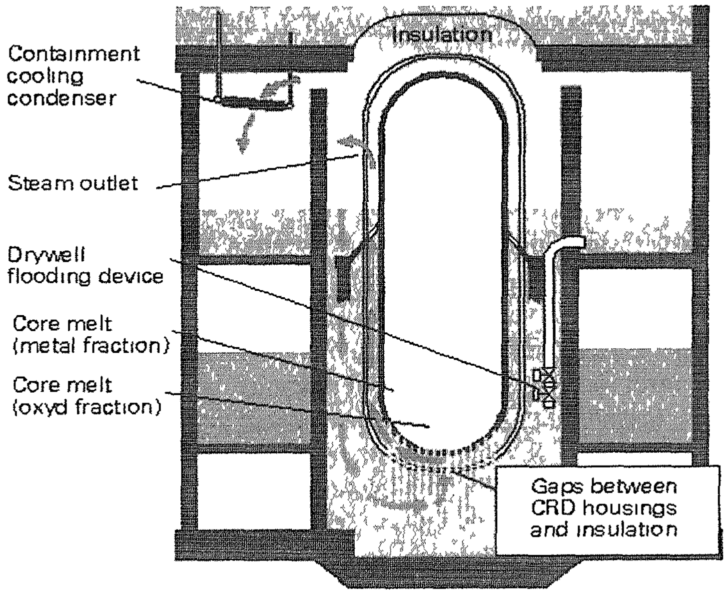

FIG. 7 Core-Melt Retention in the Reactor Pressure Vessel

3. CONTROL OF CORE MELT ACCIDENTS

Despite the much lower probability of occurrence of a core melt accident, design features are nevertheless provided for controlling an event of this kind in such a way that the consequences of the accident remain restricted to the plant and there is no need for wide-scale emergency response actions in the vicinity of the plant, such as evacuation or relocation.

Melting of the reactor core at an early point in time due to a plant-internal event involving loss of coolant is only possible if all options available for feeding coolant into the RPV should fail; in other words: both of the feedwater systems, both high-pressure pumps in the reactor water cleanup system, both low-pressure coolant injection systems and all four full-capacity passive flooding lines.

The following features guarantee control of such an event:

- The melt is retained inside the RPV by cooling the exterior of the reactor vessel. For this, a passive system is provided for supplying water from the core flooding pool to the drywell which can be activated either manually or by passive means (Figure 7).
- The high-pressure path of a core melt scenario can be practically eliminated due to the multiply-redundant and diverse equipment provided for RPV depressurization.
- An in-vessel steam explosion that could jeopardize RPV integrity if the molten material should drop into the lower plenum of the vessel cannot occur due to the internals installed in this area (control rod guide tubes).
- The containment design accounts for pressure buildup due to the hydrogen released by a 100% zirconium-water reaction of the core's zirconium inventory.
- The containment is inerted with nitrogen to prevent pressure- and temperature-raising hydrogen-oxygen reactions (deflagration or detonation).
- Retention of the melt in the RPV means that there can be no steam explosion in the containment or concrete-melt reactions with all of their consequences.

- Heat is removed passively from the containment by the containment cooling condensers to the dryer-separator storage pool outside the containment. Makeup of this pool's water, several days after the accident, ensures heat removal for an unlimited time period.
- After a few days, containment pressure can be reduced via the offgas system and its holdup filters.

Analyses performed to investigate retention of a molten core inside the RPV through cooling of the vessel exterior revealed that, based on extremely conservative assumptions, a safety factor of 4 to 5 exists with respect to the critical heat flux. When more realistic boundary conditions are assumed, the safety factor is even 10. Also, any ingress of molten material into the nozzles on the bottom head of the vessel (for the control rod drive housings and instrumentation tubes) will not cause any damage to the housings or tubes. Furthermore, the CRD housings and instrumentation tubes are supported by a special support plate that is separate from their connection to the RPV bottom head and would continue to hold them in position even if all of the internal welds attaching them to the RPV were to melt through. This concept of in-vessel melt retention for the SWR 1000 has been verified as representing a safe, reliable and inexpensive approach for controlling a postulated core melt.

4. SIMPLIFICATION OF SYSTEMS FOR NORMAL PLANT OPERATION

On the basis of past operating experience it has been possible to incorporate into the design concept of the SWR 1000 several improvements and simplifications that afford a wide variety of benefits. These include, for example:

- The active core height has been reduced from 3.7 m to 2.8 m. This allows the core to be placed lower down inside the RPV, meaning a larger coolant inventory above the core. This allows the reactor to be depressurized in the event of a LOCA down to the pressure level prevailing in the containment, without any need for coolant makeup during this period.
- The fuel assemblies will have larger, 13x13 rod arrays based on the Siemens Atrium 10 design, resulting in fewer fuel assemblies in the core as well as fewer control rods, control rod drives and in-core instrumentation assemblies.
- The long internal component parts of the control rod drives, such as the hollow piston and the threaded spindle, can be installed and removed from above through the RPV, thus eliminating the space previously needed for these activities in the control rod drive compartment underneath. As a result, the reactor vessel can be placed at a lower elevation inside the containment, allowing the overall height of the containment and the reactor building to be reduced.
- The fuel pool cooling systems previously located outside the fuel pool have been replaced by coolers installed inside the fuel pool. This eliminates an entire group of contaminated cooling systems inside the reactor building, together with their pumps and heat exchangers as well as their connections to the fuel pool.
- The fuel pool cleanup system has been combined with the reactor water cleanup system which is designed for operation under low-pressure conditions. This eliminates the filters previously provided for fuel pool cleanup (Figure 8).
- The two return pumps of the reactor water cleanup system will additionally be used for supplying cooling water to the control rod drives, seal water for the reactor water recirculation pumps and, in the event of an anticipated transient without scram (ATWS), to inject boron into the RPV.
- The number of main steam lines has been reduced from four to three, and the feedwater lines from four to two.
- The feedwater tank has been replaced by a surface-type feedwater heater.

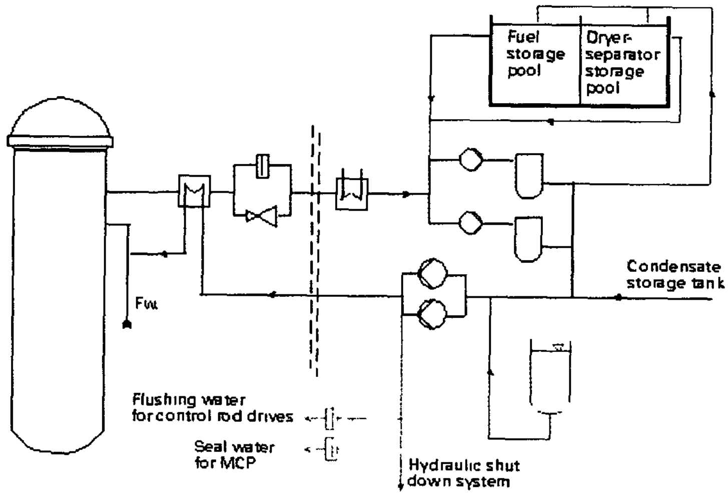

FIG. 8 Reactor water clean-up system & Fuel storage pool clean-up system

- The reheaters upstream of the low-pressure turbine sections have been dispensed with, thanks to improved stage drainage in the LP turbine sections.
- A 3000 rpm turbine generator set will be employed which is much less expensive and can fit into a smaller-sized turbine building.

These simplifications in system and component design reduce capital cost. Thus they also contribute significantly towards reducing plant construction cost as well as, of course, plant operating cost since any decrease in capital cost always means less expenditure on inspection and maintenance.

5. COMPLIANCE WITH REGULATORY REQUIREMENTS

The design of the SWR 1000 has been based on the following nuclear codes and standards:

- German nuclear regulatory codes and standards as well as recommendations issued by the Groupe Permanent Réacteurs (GPR) and the German Reactor Safety Commission (RSK) for future pressurized water reactor designs, insofar as these are applicable for the SWR 1000.
- IAEA Guidelines
- European Utility Requirements (EUR).

Furthermore, a study has shown that the safety concept of the SWR 1000 has no difficulty in meeting the Finnish nuclear regulatory requirements set forth in the YVL Guides.

In order to verify that the main aspects of the safety concept comply with German nuclear regulatory requirements, an application for a design review according to Section 7c of the Atomic Energy Act has been submitted to the pertinent authority, the Federal Office for Radiation Protection. The results of the review are expected to be available in mid-1999.

A review of EUR compliance is likewise scheduled to be carried out in 1999.

The design review activities performed so far have provided assurance that the present plant design concept meets international requirements applicable to the safety concept of an advanced reactor design.

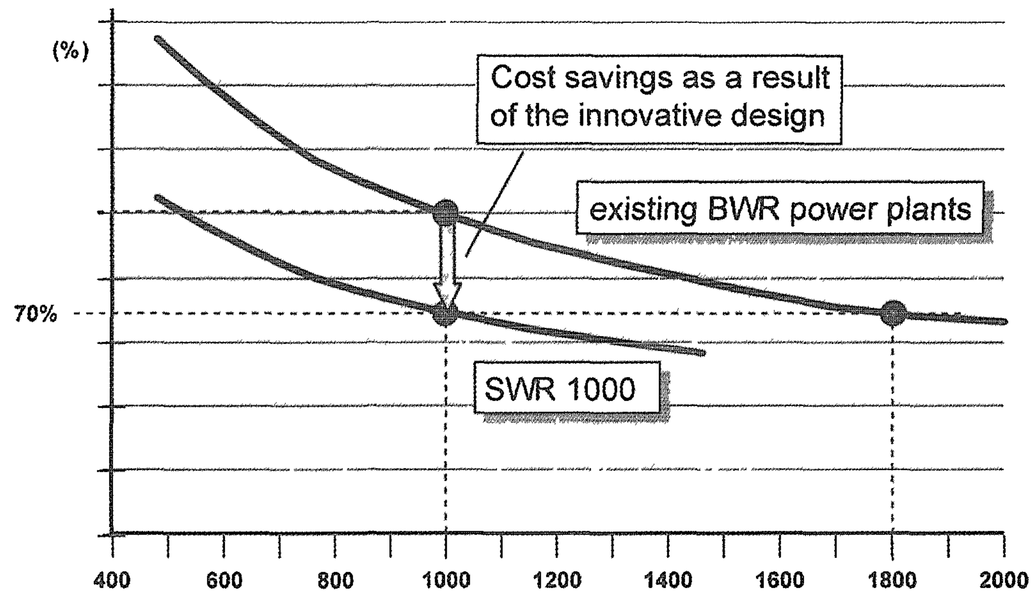

FIG. 9 Degression of the Specific Plant Cost as a Function of Power and Reactor Concept

6. COSTS AND CONSTRUCTION PERIOD

As already mentioned earlier, it has been possible to considerably reduce plant construction cost by simplifying the safety systems as well as the systems needed for normal plant operation. A detailed comparison of the plant construction cost of the SWR 1000 with that of a traditional, contemporary BWR plant of the same capacity has revealed that the SWR 1000 costs 30% less; i.e. the cost of building the SWR 1000 amounts to only 70% of the cost of constructing a traditional BWR plant of the same size (Figure 9). Thus, the specific cost for turnkey construction of a nuclear power plant equipped with an SWR 1000 (excluding the owner's own direct contribution) amounts to around 2500 DM/kW (or, according to present exchange rates, 1470 US$/kW).

As plant simplification also leads to a reduction in operating cost, such as expenditure for in-service inspections and maintenance, as well as to a need for fewer personnel for maintenance and repair work, the power generating cost is altogether lower than that of other nuclear plants with similar fuel cycle costs. In fact, assuming an annual operating time of 7500 hours, an amortization period of 30 years, pre-tax returns of 11% and a financing package with two-thirds of the capital borrowed at an interest rate of 7.5%, the power generating cost of an SWR 1000 is around 5.5 pfennigs, or 3.2 US cents, per kilowatt-hour.

The smaller interior building volumes, the reduced scope of components and equipment, and the use of state-of-the-art construction methods all result in a short construction period of just 48 months.

7. STATUS OF DEVELOPMENT AND COMMERCIAL MATURITY

The SWR 1000 development project is currently in its Basic Design Phase, which started in mid-1995 and will conclude in mid-1999 with the issuance of a safety analysis report and an estimate of construction costs. In addition to this development work, which also includes a design review for regulatory compliance, all new passive safety components have been successfully put through full-scale tests on various test rigs in Switzerland and Germany to verify their functional reliability and capacities. At the moment, the passive flooding system is still undergoing tests at the Jülich Research

Center and the effects of aerosol deposition on the containment cooling condenser are presently being investigated as part of the CONGA Research Program of the European Union. Another series of tests, aimed at verifying the results of analyses carried out of cooling of the RPV exterior in the event of a core melt accident, is still at the planning stage.

The present status of development and the results obtained from experimental verification of the new systems and components show that the design concept of the SWR 1000 will reach commercial maturity by the year 2000.

KEY DEVELOPMENTS OF EVOLUTIONARY DESIGNS

(Session IV b)

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Hungary

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Republic of Korea

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KEY DEVELOPMENTS IN THE ADVANCED NPP WITH WWER-640/V-407 REACTOR PLANT DESIGN

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Abstract

The report covers the main design features of advanced NPP equipped with WWER-640 reactor, that take into account the up-to-date approaches in the process of forming safety concepts. An approach to accident management has been analysed, beyond design-basis accidents included. A description of principal safety systems has been presented as well as the interrelation of their operation. The principal features of the systems design have been shown.

1. INTRODUCTION

WWER-640 reactor (Model V-407) is a vessel-type pressurized water reactor. The majority of reactor plant components are similar to the equipment that has shown a good performance at the operating NPPs and has been adopted for commercial manufacturing for WWER-440 (model V-213) and WWER-1000 (model V-320). The NPP safety systems have been designed as passive systems that do not require any operator actions within 24 hours. Up-to-date Russian approaches have been implemented and world experience has been used in determining safety levels and their relation with the safety barriers in the process of forming safety concept.

2. LICENSING STATUS

The procedure of licensing has been envisaged in the acting regulatory documents of the Supervisory bodies, namely, of RF Gosatomnadzor.

The requirements for granting a license in Russia comprise the procedure of step-by-step project licensing, up to NPP commissioning for commercial operation. They are given in [1, 2].

Currently siting permits and NPP construction licenses have been granted for the Kola-2 site and Nuclear Research and Industrial Centre in Sosnovy Bor.

3. REGULATORY BASIS FOR ADVANCED NPP WITH WWER-640 DESIGN

The advanced NPP equipped with WWER-640 was designed in accordance with the regulatory documents and standards enacted in the Russian Federation as well as with allowance for the IAEA Safety Guides. The analysis was carried out of design compliance with a number of principal regulatory documents, applied in the USA, Germany, France. This analysis has shown that design safety can be confirmed on the basis of regulatory documents from any of these countries.

The regulatory requirements are met in the design along with an acceptable level of economic indices due to:

(1) Double containment, application of passive emergency core cooling systems and passive residual heat removal systems in the design, which have reduced the probability of severe accidents by order 2 or 3 in comparison with WWER-1000 (Model 320) [3];

(2) The fuel cost decrease through improving nuclear fuel cycle, which leads to an increase in fuel utilisation efficiency by 20-25%;

(3) Upgrading of automation and process control systems as well as furnishing them with equipment and pipelines diagnostics systems;

(4) Decrease of the amount of equipment and its total quantity of metal;

(5) Service life elongation up to 50-60 years.

4. DESIGN EVOLVING FROM THE POINT OF VIEW OF SAFETY ASSURANCE BY MEANS OF ACTIVE OR PASSIVE SAFETY SYSTEMS

The traditional demand for meeting the increasing safety needs in new NPP designs is satisfied by safety systems growth and their sophistication. This tends to result in an increase in the total cost of the Unit. Such an increase was in previous designs partly compensated for by increasing the unit power. Sophistication of the safety systems yields more complicated servicing and control, which increases the risk of human error.

For this reason, two approaches to design of new generation PWR type reactors are being simultaneously pursued at present:

- the first approach is based on the development of large power plants with the prevailing usage of active safety systems;
- the second approach is based on creating medium-size power reactors with a preferred usage of passive safety systems.

The reactor plant with WWER-640 belongs to the second approach. In the WWER-640, the significance and quantity of active systems are reduced to a minimum. In reality, there is only one safety system that is of active type, namely, the system of boron supply with high-pressure pumps in case of an ATWS-type accident. The other active systems are only required for normal operation.

The choice of systems, operating on the basis of passive principle is stipulated by several causes:

(1) A possibility to use natural phenomena forces in principle, proceeding from the correlation of the Unit scale factors and its thermal power;

(2) Engineering and economical expediency;

(3) An attempt to exclude operator's errors that can arise, for example, as a result of restoring one safety function or another.

5. PRINCIPAL GOALS AND METHODS TO IMPLEMENT SAFETY CONCEPTS

The following principal goals have been established for the safety concept of the NPP design with WWER-640:

- not less than one passive safety train shall be envisaged;
- time interval before the necessity of operator's actions to restore or maintain the principal safety functions shall be not less 24 hours;
- the content of the activities shall be aimed at safety functions restoring by simple make-up or start-up of available systems;
- an additional safety barrier shall be envisaged in case of a severe (beyond design basis) accident involving core melting.

The following technical provisions are envisaged in the reactor plant design concept with WWER-640 to implement the principal goals of the concept:

(1) Decreased power density in the core;

(2) Lower neutron flux onto reactor vessel;

(3) A higher efficiency of the system of changing reactivity due to increasing the number of CPS mechanical members;

(4) A large inventory of reactor and pressurizer coolant;

(5) Passive coping with an accident involving loss of all the sources of AC power without any operator's actions within 24 hours;

(6) Elimination of transients without any need of emergency make-up;

(7) Passive features for realisation of safety system functions; large inventories of borated cooling water inside the containment, furnished with facilities providing water flow into reactor vessel at pressure decrease inside the latter;

(8) Alternative systems of heat removal from reactor vessel and the containment ;

(9) Systems of removing hydrogen from containment;

(10) A possibility of passive external heat removal from reactor vessel;

(11) Advanced containment for elimination of core melt accidents.

6. DESIGN BASES FOR ACCIDENT MANAGEMENT CONCEPT

The principle of implementation of defence-in-depth is the basis of safety concept. The principle is realised by the usage of:

- a system of physical barriers on the way of ionising radiation propagation and radioactive substances release into the environment;
- engineering and organisational measures intended to protect the barriers and maintain their efficiency as well as direct measures to protect the population.

The system of physical barriers is traditional for NPP with WWER reactor and comprises, respectively:

- fuel matrix;
- fuel element cladding;
- primary coolant pressure boundary;
- envelope.

A system for retention and localisation of non-gaseous fission products that can be released in case of reactor vessel destruction is envisaged as an intermediate safety barrier between primary circuit boundary and the envelope. Elaboration of possible engineering solutions for this system is currently under way. Meanwhile, at least the necessary space of reactor cavity is provided. The space is intended to catch and safely confine corium. With this aim appropriate post-accident monitoring means are envisaged in the cavity space.

In accordance with [4] the system of engineering and organisational provisions is to create five levels of defence-in-depth. With this, each level of engineering and organisational provisions is assigned appropriate levels of the NPP status that characterise the conditions of power unit operation. The purpose of splitting out such horizontal links is to prevent NPP transition from a higher level to a lower one or providing purposeful and efficient reactor plant protection in case

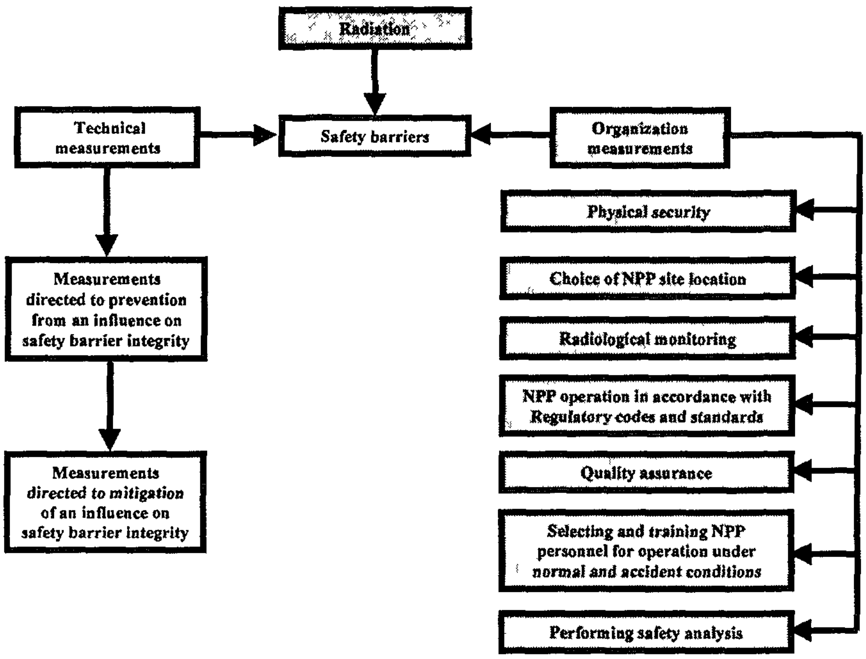

FIG. 1. The application of the defence-in-depth principle

such a transition has taken place. Figure 1 illustrates the application of the defence-in-depth principle.

Safety functions have been determined in the project in order to provide the accomplishment of the purpose of safety barriers integrity protection and its efficiency. It meets the requirements of the up-to-date regulatory documents [4, 5]. For each of the safety barriers to retain their integrity a number of conditions exists that must be permanently maintained. These conditions are ensured by means of safety functions, taken from the general list of safety functions.

7. REALIZATION OF SAFETY BARRIERS AND SAFETY FUNCTIONS IN THE DESIGN OF NPP WITH WWER-640 REACTOR

Design solutions on engineering provisions aimed at supporting the efficiency and integrity of safety barriers, enumerated in item 6, will be further analysed.

7.1 Fuel Rod Claddings and Fuel

Reactor core has been developed on the basis of the operating NPPs with WWER-1000 reactors. As a result of it a core with a lower power density was created. Safety analyses show that the fuel rod cladding temperature will not exceed 800 °C under all design basis accidents, that the fuel rods will be under these conditions for not more than 150 s, and that the local depth of oxidation does not exceed 5%.

With this, fuel assembly design has been improved due to:

(1) Application of progressive structural materials;

(2) Providing fuel assemblies repairability;

(3) Usage of gadolinium as burnable poison;

(4) Fuel enrichment radial profile shaping within the fuel assembly

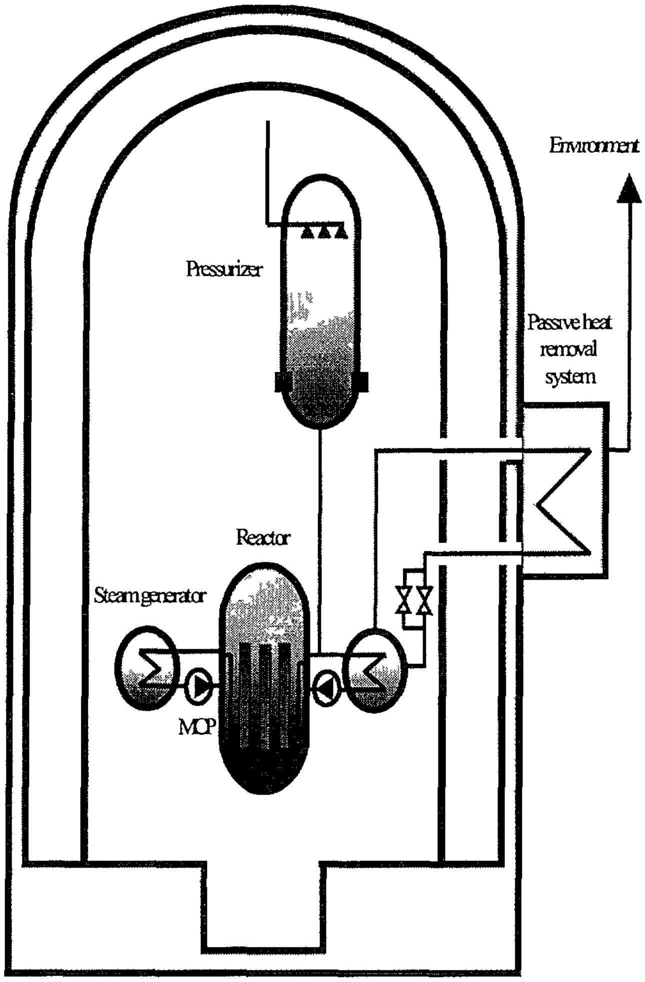

FIG. 2. The operation of passive heat removal systems under non-LOCA conditions

7.2 Primary Pressure Boundaries

Circulation loops are not equipped with loop seals and consist of straight tubes and bends in the places of connection to steam generators and pumps.

Main circulation circuit equipment layout as well as passive heat removal equipment layout provides residual heat removal from the core into the tanks of chemically demineralized water and further to the atmosphere following reactor shutdown by natural circulation. Reactor power that can be removed from the core by coolant natural circulation on the basis of the calculated data is 10% of the nominal value, which guarantees a reliable residual heat removal.

Thus, in case of accidents not involving primary coolant loss following reactor shutdown by appropriate signals of emergency protection system, heat is removed from the core, as shown in Figure 2, due to primary coolant natural circulation to steam generator boiler water. The steam generated comes into the steam generator passive heat removal system (PHRS). In the SG PHRS steam is condensed on the internal surface of the tubes that are cooled on the outside with water stored in the demineralized water tanks. The condensate then flows back into the steam generators by gravity.

7.3 Containment

A double protective and confining containment arrangement is provided in the design of NPP equipped with WWER-640 reactor.

The primary hermetic envelope is made of steel.

The secondary envelope, the containment proper, is made of concrete. This containment is designed to protect reactor from external impacts, namely:

- aircrash,
- shock wave.

Small under-pressure is maintained inside the gap between the envelopes and exhaust air passes through filters into the ventilation tube.

The outer containment is separated from the steel envelope and it does not transfer the loads produced by any external impacts to the steel envelope. Both envelopes are supported by the reactor building bedplate and fixed to it.

All the primary equipment and pipelines are housed inside the reactor hall hermetic envelope, and all the connections pass through hermetic penetrations, that prevent pressurized coolant release from the envelope under any accident situations. The system of passive ECCS is also housed inside the envelope.

The reactor plant is so located that in case of an accident involving considerable primary coolant loss, and actuation of the ECCS, the reactor plant is flooded together with the fuel pond, which provides reliable cooling of all the fuel inside the envelope. Thus, the concrete pit where the reactor is located, the floor of the steam generators' compartment and the compartment walls form an emergency heat removal pond.

Primary pressure is decreased in case of loss-of coolant accidents by means of the passive heat removal system and by mass and power release into the envelope. Following primary pressure decrease to 4-6 MPa, check valves open at the ECCS tanks and boric solution begins flowing into the reactor vessel. The further cooldown and pressure decrease in case of small-break and medium-break LOCAs are realised via steam generator PHRS and hermetic envelope PHRS.

When the primary circuit and hermetic envelope pressure differential has decreased to 0,6 MPa, the passive valves of the emergency depressurization system, which connect the loops hot and cold legs with the fuel ponds space, open.

The opening of the depressurizing valve will yield further pressure decrease. When the primary circuit and ECCS tanks pressure differential has decreased below 0,3 MPa, the tanks begin reactor flooding.

This results in establishing a natural circulation along the following flow path, as shown in Figure 3: reactor lower plenum - core - reactor upper plenum - "hot" leg - pipeline with valve system for depressurization - fuel pond - pipeline with valve system for depressurization - "cold" leg - reactor lower plenum. The given flow path provides long-time heat removal from the core in case of LOCAs combined with loss of all electric power.

As the level in the emergency pond increases (approximately up to the level of the "hot leg" nozzles on the reactor vessel), the valves of connecting line between the emergency pond and the fuel pond open. As a result, the entire inventory of water stored inside the steel envelope will become involved in the process of cooling the core and the spent fuel.

The integration of the ponds leads to water mixing with a mean temperature below the saturation temperature. The water of both ponds continues getting heated, and in about 10 hours the water reaches boiling temperature. The generated steam condenses on the internal surface of the hermetic envelope, and the condensate flows back into the pond. This steam condensing is

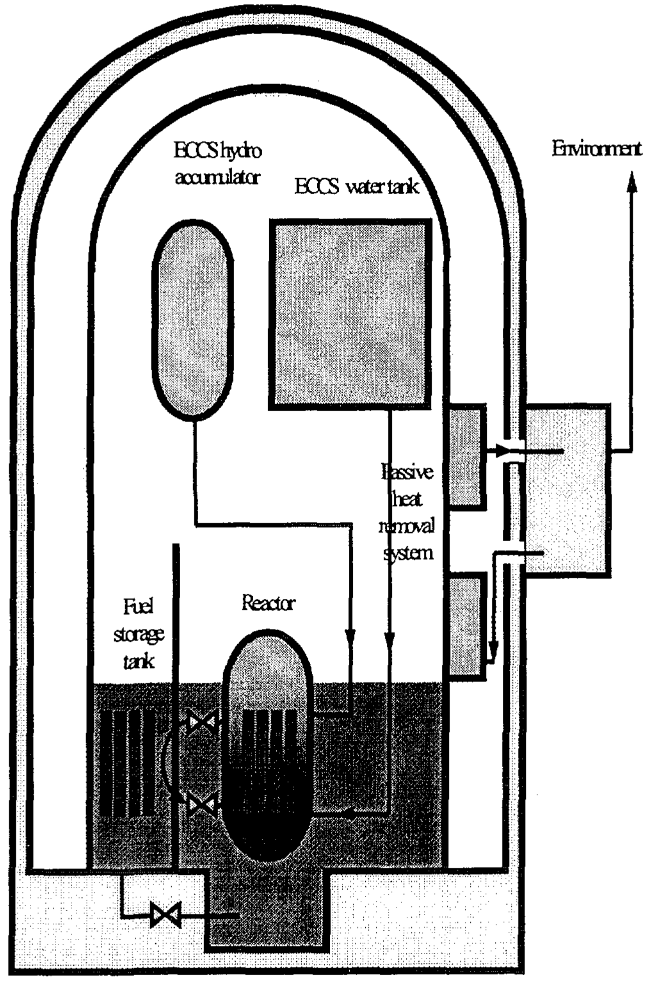

FIG. 3. The operation of passive heat removal systems under LOCA conditions

provided by the system of hermetic steel envelope cooling (HE HRS), which is designed for 24 hours operation without any operator actions.

Long-time heat removal from the core is thus provided in case of loss-of coolant accident and loss of all power.

8. CONCLUSION

As a result of assuming the decisions, solving the following safety-related problems is provided in the design:

(a) Reliable cooldown and long-time core heat removal (for 24 hours) without any operator actions is provided;

(b) Control and protection system keeps the core sub-critical at any moment of the fuel cycle under any loss-of-coolant accident even in case of clean condensate supply;

(c) Passive reactor core flooding on the outside under any loss-of-coolant accident to remove heat from reactor vessel bottom by natural convection in case of postulated core destruction and corium agglomeration on the vessel bottom.

List of Adopted Abbreviations

VSP	- valve system for depressurization
WWER	- water-cooled and water moderated power reactor
SG	- steam generator
ECCS	- emergency core cooling system
HE HRS	- hermetic envelope heat removal system
PHRS	- passive heat removal system
CPS	- control and protection system
SF	- safety function

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THE NEXT GENERATION CANDU 6

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Abstract

AECL's product line of CANDU 6 and CANDU 9 nuclear power plants are adapted to respond to changing market conditions, experience feedback and technological development by a continuous improvement process of design evolution. The CANDU 6 Nuclear Power Plant design is a successful family of nuclear units, with the first four units entering service in 1983, and the most recent entering service this year. A further four CANDU 6 units are under construction. Starting in 1996, a focused forward-looking development program is under way at AECL to incorporate a series of individual improvements and integrate them into the CANDU 6, leading to the evolutionary development of the next-generation enhanced CANDU 6. The CANDU 6 improvements program includes all aspects of an NPP project, including engineering tools improvements, design for improved constructability, scheduling for faster, more streamlined commissioning, and improved operating performance. This enhanced CANDU 6 product will combine the benefits of design provenness (drawing on the more than 70 reactor-years experience of the seven operating CANDU 6 units), with the advantages of an evolutionary next-generation design. Features of the enhanced CANDU 6 design include:

- Advanced Human Machine Interface - built around the Advanced CANDU Control Centre.
- Advanced fuel design - using the newly demonstrated CANFLEX fuel bundle.
- Improved Efficiency based on improved utilization of waste heat.
- Streamlined System Design - including simplifications to improve performance and safety system reliability.
- Advanced Engineering Tools, -- featuring linked electronic databases from 3D CADDS, equipment specification and material management.
- Advanced Construction Techniques – based on open top equipment installation and the use of small skid mounted modules.
- Options defined for Passive Heat Sink capability and low-enrichment core optimization.

1. INTRODUCTION

AECL's CANDU 6 NPP product line is based on the foundation of a successful series of operating power plants and current build projects. Using the continuous improvement approach, incremental design improvements have been incorporated in successive projects. By extending this approach to incorporate the benefits of AECL's product development programs, a series of improvements can be mapped out for the future, leading to a next-generation enhanced CANDU 6.

In practice, AECL's approach for CANDU 6 is to incorporate design improvements based on: provenness; incremental change; benefit to plant performance and safety; benefits to plant economics. Design changes are thoroughly evaluated to ensure these objectives are achieved. As developments meet the criteria of provenness and readiness for implementation, they can be seamlessly incorporated into the next CANDU 6 build project, offering customer benefits with very low implementation risk. This approach has already been followed in design upgrades for current projects at Wolsong, in the Republic of Korea, and Qinshan, in China.

In addition, AECL's design evolutionary approach emphasizes systematic design response to feedback from operating plants, build projects, from R and D programs and from worldwide experience. In this way, the design is maintained up-to-date and is fine-tuned to minimize operating problems. Design "provenness" is maintained, not by freezing the design, but by a continuous, "living" design improvement process.

The CANDU 6 design has led to a successful family of units, with the first four units entering service in 1983 (Point Lepreau and Gentilly-2 in Canada; Embalse in Argentina; and Wolsong-1 in the Republic of Korea. More recently, further CANDU 6 units are under construction, and three units have recently entered service (Cernavoda-1 in Romania, in 1996; Wolsong-2 in 1997; and Wolsong-3 in 1998). In addition, units are under construction at Cernavoda-2, Wolsong-4 and Qinshan 1 and 2.

In 1996, a focused CANDU 6 improvements program was started, with the objective of identifying and developing successive design improvements to establish the next generation, enhanced CANDU 6 design. This program includes all elements of an NPP project in its scope. This includes not only design improvements, but also engineering tools improvements, design for improved constructability, scheduling for faster, more streamlined commissioning, and improved operating performance. This enhanced CANDU 6 represents the outcome of the natural evolution of the CANDU 6 design, incorporating design feedback in conjunction with the design improvements from this development program. The enhanced CANDU 6 NPP design combines the benefits of design provenness (drawing on the more than 70 reactor years experience of the seven operating CANDU 6 units), with the advantages of an evolutionary, next generation design. The principles behind AECL's approach to evolutionary improvement are shown on figure 1.

In this paper, examples are given of the key incremental design features included in successive CANDU 6 NPP's together with a summary description of the improvements in the next-generation enhanced CANDU.

2. CONTROL CENTRE DESIGN EVOLUTION

The original CANDU 6 control centre and human-machine interface incorporated a high degree of automation and digital control. From the beginning, the CANDU 6 design philosophy emphasized provision of extended operator decision and action time for upset and accident conditions.

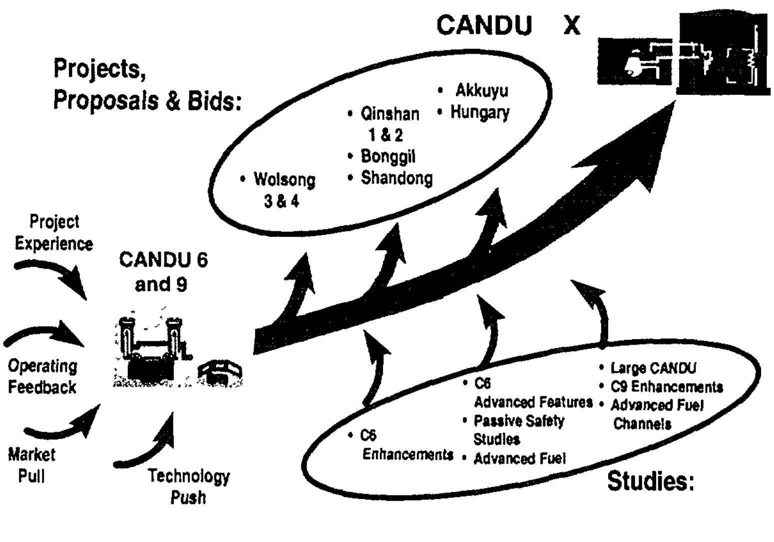

FIG. 1 CANDU Product Evolution

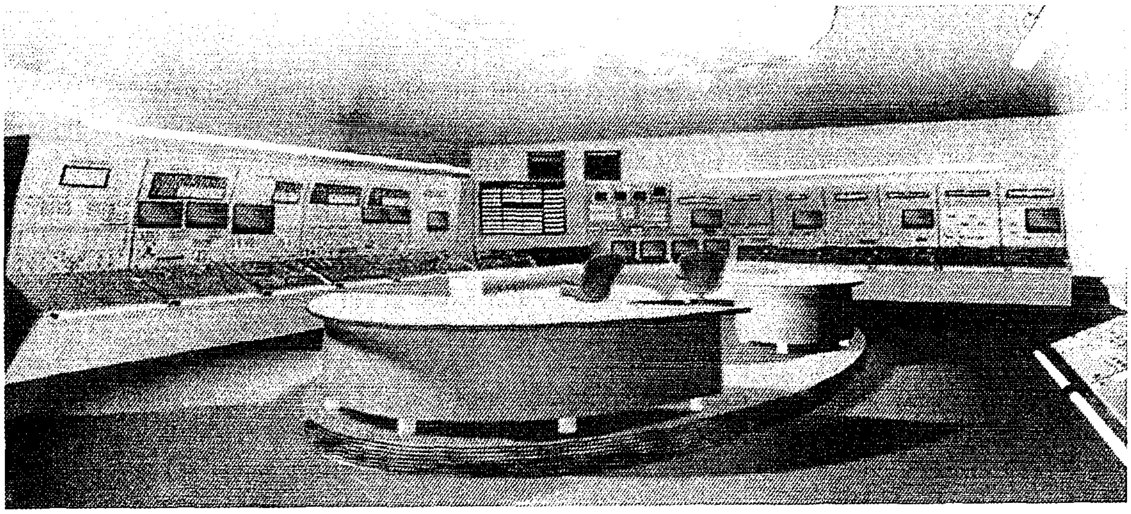

FIG. 2 Advanced Control Room

The original control centre design for the CANDU 6 plants completed in the 1980's, included plant control and display functions carried out by dual-redundant computers, with extensive use of CRT displays. Refinements for the Wolsong 2, 3, and 4 projects included redesigned emergency core cooling system (ECCS) panels, to reflect the complete automation of all stages of ECCS operation. For Qinshan, the Safety Parameter Display System has been integrated into the design, and the control room panels have been simplified and re-oriented based on human factors principles. Most notably, the operator interface has been streamlined by inclusion of extensive CRT-based control and information display at the central operator sit-down console, and by the inclusion of the advanced CANDU messaging system to provide "intelligent" annunciation during upset conditions. Operator situation awareness and analysis is enhanced by the use of two large-screen plant overview displays.

The next CANDU 6 projects will implement the Advanced CANDU Control Centre (Figure 2). In addition to the enhancements noted above, this will feature:

- Real time plant information, and historical data, communicated throughout the plant via Local Area Network.
- Extended automation of safety system on-line testing and calibration.
- Touch-screen context-sensitive CRT's as the first-line operator interface.

These improvements respond to utility expectations to have an operator interface based solidly on human factors principles, and which simplifies operator management of accidents Reference (1).

Finally, the future enhanced CANDU 6 will adopt plant control by a distributed microprocessor based system for both NSSS and BOP equipment. This is readily incorporated due to the separation between the digital control and display functions in the control equipment architecture. The use of multiplexed distributed control, in combination with the separation of control and display functions, will greatly reduce the effort and schedule duration required to complete wiring during project construction. For example, this allows the elimination of the complex Control Distribution Frame of the original CANDU 6 units, with resulting savings in on-site critical path wiring activities.

3. FUEL DESIGN EVOLUTION

The CANDU 6 reactor core is designed around the highly-proven, economical 37-element CANDU fuel bundle. The CANDU 6 core design allows fuelling with a variety of low fissile-content fuels, in particular natural uranium. Up to the present time, the natural uranium fuel cycle has been used in all CANDU 6 units, to take advantage of its simplicity, and of the independence from fuel enrichment. 37-element CANDU fuel has achieved excellent performance in CANDU 6 and other CANDU power plants. The average fuel defect rate is less than one in 100,000 fuel pins. Because CANDU plants refuel on-line, any fuel defect which does occur can be located and removed promptly. This contributes to the very low activity levels typically seen in CANDU coolant circuits.

AECL has carried out a development program with KAERI of Korea, to make available an advanced 43-element fuel bundle, (Figure 3), the CANFLEX fuel bundle, which will enable a flexible choice of fuel cycles to suit utility strategy. By the choice of two carefully selected fuel pin diameters in the fuel bundle, the CANFLEX bundle achieves 20% lower fuel ratings than 37-element fuel, based on the same overall bundle power. This leads to lower fuel temperatures and free fission-gas release, which allows much higher burnups, and hence opens up the competitiveness of alternate fuel cycles. In addition, the CANFLEX fuel bundle is designed to achieve significantly higher thermal margin, which gives flexibility to optimize the core design, while providing greater flexibility to the plant operators to vary core conditions.

Demonstration irradiation of CANFLEX fuel bundles in a CANDU 6 reactor began in September of this year at Point Lepreau NGS in New Brunswick, Canada. After successful irradiation and post-irradiation of the demonstration bundles, transition to full-core CANFLEX fuel at Point Lepreau is anticipated.

The present CANDU 6 design is optimized for 37-element fuel. The enhanced CANDU 6 plant includes the same core design, with flux detector configuration adjusted to optimize the reactor for CANFLEX fuel. The reference fuel management scheme and fuel channel flow designs are also adjusted to complement this, allowing up to 6% increased thermal output from a natural uranium-fuelled core, while establishing higher thermal margin for greater operational flexibility. The future CANDU 6 design will allow the option of power plant optimization for slightly enriched uranium fuel (0.9% to 1.2% b.w. U-235), to permit a further 5% output increase, in combination with significantly extended burnup (a factor of 2 or more higher). The CANDU 6 core design is readily adapted to optimization for variations in fuel cycle (in fact, existing CANDU 6 plants can be fuelled with slightly enriched fuel with no back-fit requirements except flux detector re-calibration). Therefore such designs will provide future opportunities to take advantage of fuel cycle options for the future to reduce fuelling cost, improve uranium utilization, or make maximum use of alternate fuel sources

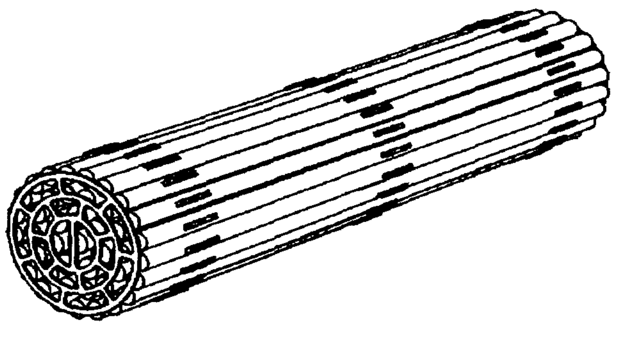

FIG. 3 CANFLEX Fuel Bundle

such as Recovered Uranium (RU), MOX, or thorium. Reference (2) describes the range of fuel cycle options which can be considered for CANDU in the future.

4. SAFETY DESIGN EVOLUTION

The CANDU 6 has, from the start, included safety design features which are familiar as standard parts of the CANDU family of designs:

- Two separate, diverse, fully redundant shutdown systems.
- All reactivity devices contained in the cool, low-pressure moderator.
- All safety systems designed for on-line testing to demonstrate 10^{-3} system unavailability continuously during operation.
- Moderator D_2O available as an emergency heat sink in the event of coolant loss coupled with assumed failure of emergency core cooling (ECC).

As the CANDU 6 design has evolved, the fundamental approach to safety design has not changed, but incremental improvements have been made. Incremental changes so far include:

- Automation of transfer from medium-pressure to low-pressure ECC, establishing full automation of all ECC functionality.
- Use of qualified, licensed, safety-critical software for digital control and testing of each shutdown system.
- Duplication of ECC heat exchanger and key isolating valves for easier testability and greater demand availability.
- Separate safety system monitoring display computer and console in Control Room.
- Dual-redundant, seismically-qualified fire suppression sprays.

Features under development for the enhanced CANDU 6 design include:

- Improved fuel thermal margin through the use of CANFLEX fuel and improved core flow and temperature monitoring.
- Reactor shielding vault cooling system design optimized to enhance inherent emergency heat sink capability for severe core damage accidents.
- Simplified, high-pressure design containment, eliminating requirement for fast-acting dousing pressure suppression sprays.
- Emergency core cooling system gas / H_2O / D_2O isolation using simplified, passive components.
- Increased redundancy and operating flexibility of both auxiliary and emergency feedwater systems.

5. POWER SYSTEMS DESIGN EVOLUTION

The CANDU 6 Primary Coolant System and associated systems use a well-established equipment configuration (2 coolant loops, each with two steam generators and two coolant pumps). Detail improvements to power systems design, implemented in current plants, include:

- Improved fuel channel piping materials, with higher chrome content to enable 60-year effective operating life.

- Steam generator design including improved access for inspection and cleaning.
- Fuel Channel design with greater seismic margin for more site flexibility, and featuring pressure tube design and material selection for longer life.
- Moderator system design with optimized core D_20 flow distribution, to allow higher temperature operation and save on heat sink equipment cost.

The CANDU 6 product optimization program has established a series of design improvements to improve performance, operating life, and safety margin:

- Moderator waste heat recycle via low-pressure feedwater heating, allowing over 1% improvement in output while achieving equipment cost reductions.
- Heat transport system purification optimized for precise control of pH and related chemistry parameters.

In addition, important improvements have been included for the next CANDU 6 in design of plant service systems. For example, the reactor building and service building ventilation and vapour recovery systems have been simplified and optimized so as to reduce emissions of tritium by a factor of four in comparison to the original CANDU 6 units.

Similarly, the use of the AECL-developed CECE catalytic exchange process improves effectiveness of heavy water in-house upgrading.

6. ENGINEERING METHODS EVOLUTION

Recent CANDU 6 build projects have incorporated successive advances in the use of advanced engineering tools. The Wolsong 2, 3 and 4 projects were based on use of 2-D CADDS to produce a reference plant design model through a series of linked CADDS databases. This enabled successes in project implementation through reduced design rework and field changes. This evolution has continued to the Qinshan project where a full 3-D CADDS model is used as the basis for CANDU 6 design. Electronic data transfer to engineering tools for such activities as support design, and stress analysis, has streamlined project execution. One result is a dramatic reduction in the number of field interferences, with benefits in reduced installation rework.

In wiring design, AECL introduced the InTEC wiring software for the Wolsong 2, 3, 4 project. This software links the design, engineering, procurement and installation processes.

For future CANDU 6 projects, AECL's suite of software tools such as 3D CADDS will be fully linked, with the inclusion of key functions of equipment specification (through TeddyBase the Tagged Equipment Database, and CMMS project (CANDU Material Management System), along with intelligent databases for document and other deliverable control.

At this point, AECL has developed, and gained real-world project experience with, a comprehensive suite of engineering tools, which enable efficient, timely generation and communication of engineering information throughout the project team, and throughout the duration of project execution.

7. PROJECT DELIVERY EVOLUTION

Project delivery for CANDU 6 units has also been improved through a continuous process of evolution. In particular, the use of open-top construction with equipment installation using heavy lift cranes, and the application of small, skid mounted modules, will streamline and shorten total project schedules. The Qinshan CANDU 6 project is currently using open-top methods extensively for major equipment installation. Future CANDU 6 projects will build on this experience to enhance the application of open top techniques. In addition the CANDU 6 improvements program has identified and designed 8 selected system modules which can be pre-fabricated and lifted into position "over-the-top", in addition to structural modules and rebar prefabrication. These delivery improvements will lead to further project schedule reductions. Currently CANDU 6 projects have achieved on-time completion in as low as 69 months from Contract Effective Date (CED) to in-service (Wolsong-3). For the immediate future, a target 66 month schedule has been established.

8. OPERABILITY EVOLUTION

The advancement of plant control, instrumentation and monitoring (section 2 above) is part of an evolution towards making operations and maintenance simpler, less time-consuming and more effective. This is recognized to be a key utility desire, and is an objective in all areas of design improvement. It is considered both for hardware design and for the provision of support information, for example linking operations and maintenance manuals to design basis and plant detailed configuration information.

The comprehensive, linked use of engineering tools, as described in section 6 above, allows a much more accurate, organized, and easily retrievable body of operating support information. The inclusion of plant operational history storage, available on the station LAN to both operations, helps maintenance and technical support staff to make effective maintenance planning decisions. AECL is carrying out studies in partnership with CANDU utilities to identify how this improved information availability can be used to generate and implement Condition-Based Maintenance (CBM) procedures for key systems. Condition Based Maintenance optimizes preventive maintenance and inspection plans, to reduce maintenance staff burden while at the same time improving equipment reliability. In support of this, additional Equipment Status Monitoring instrumentation is being included in the design for key plant equipment. In addition, AECL's development programs have established and tested equipment improvements in instrumentation self testing and signal noise analysis, which improve equipment status knowledge.

9. PASSIVE HEAT SINK CAPABILITY FOR CANDU 6

The CANDU 6 plant design has emphasized the use of passive systems, features and components to maximize reliability. Safety systems such as the two shutdown systems and the emergency core cooling systems use stored energy from high-pressure gas, or gravity, as part of the poised state, requiring the minimum of signal and valve response to act. The fully automated emergency core cooling together with the design for dual redundant trips on each shutdown system for all events, means that for design basis events, the enhanced design will allow an operator a grace period of ~ hours before safety-critical action is required.

AECL's development program has included several years of conceptual and detailed design and proof testing of further passive heat sink features. The objective is to take advantage of these features to improve system reliability, further simplify operator response and improve maintainability by minimizing the safety-critical requirements for active components and support systems. The introduction of passive features is planned as an evolutionary process. It is not considered necessary to define systems as fully "passive" or "active". The benefits of passive features are measured in overall design outcomes such as system availability on demand, long term reliability, or degree of dependence on operator action. Passive heat sink features from this program are scheduled to be included in the longer term development of the CANDU 6.

The AECL program of passive heat sink development is described in Reference 3. The conceptual design of a linked group of passive heat sinks for CANDU 6 is shown in figure 4.

The centrepiece of the passive heat sink approach is the Passive Emergency Water Supply (PEWS) tank, a porous-shaped tank located at a high elevation in the reactor building.

The basic requirement is to supply cooling water for decay heat removal for an extended period, and the PEWS tank can supply water for various systems and functions, dependent on the initiating event and whether any subsequent system failures have occurred. The two heat sink designs planned for early implementation in the CANDU 6 include the use of natural circulation moderator cooling (enhanced by forced circulation at full reactor power) and natural circulation containment atmosphere cooling. In both cases, the heat sink is a passive backup to the existing engineered system, Emergency Water Supply (for failures of normal steam generator heat removal).

10. DESIGN OVERVIEW FOR ENHANCED CANDU 6

The CANDU 6 Improvements Program is based on developing and implementing a series of incremental improvements while maintaining the proven nature of the design. In this way the project risk of implementing change is kept low. Also there is strong assurance that the good operating performance shown by the existing CANDU 6 plants will be repeated or exceeded. Each improvement is selected to be practical when implemented on its own, but also to complement other planned design changes.

AECL's design direction is based on responsiveness to utility and regulatory requirements. For example, the CANDU 6 design has been reviewed in comparison to the EPRI Utility Requirements Document, and found to meet the high-level requirements (the URD was developed principally for LWR's, so one-for-one comparison at a detailed level is not practical). The CANDU 6 development direction is to maintain compliance with the principles of major utility requirements, and to respond to utility expectations, such as the use of human factors principles in the human-machine interface design. Similarly, with regard to regulatory expectations, the direction is to maintain and enhance the ability to carry out timely plant licensing in a diverse range of national jurisdictions. The enhanced CANDU 6 design will continue to comply with IAEA safety guides and with evolving Canadian

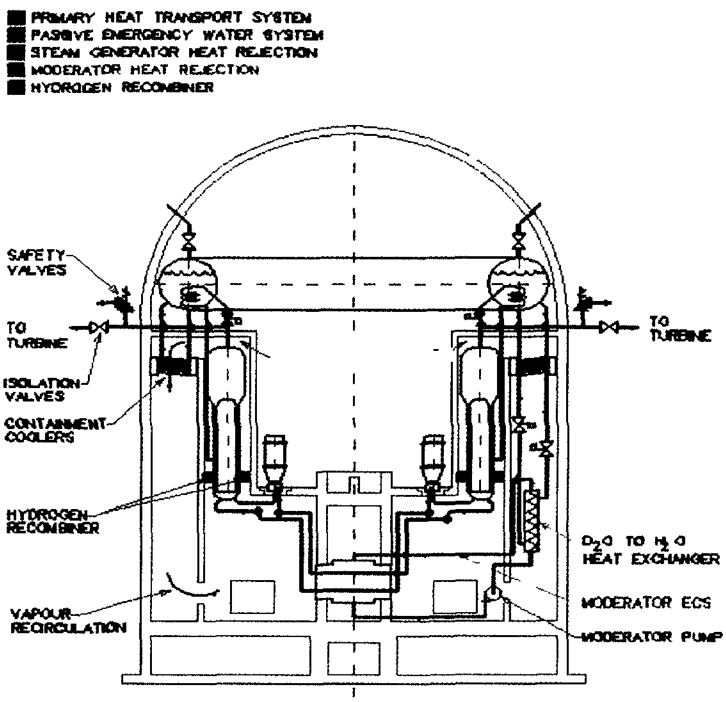

FIG. 4 Illustration of passive CANDU 6 system

licensing regulations, and will maintain a continuous link with established licensing approvals, for example regarding safety critical software methods.

The process of improvement selection, definition and implementation follows a careful review path establishing that individual changes are: proven (development testing is complete); incremental (limited cascading effect on rest of plant); beneficial to performance (e.g. capacity factor improvement; lower maintenance burden; lower risk of operator error, etc); beneficial to safety (improved margins; improved safety-critical system reliability); and / or beneficial to economics (reduced supply cost, construction cost, schedule, operating cost). In this way, change implementation occurs with very low risk to the customer.

The outcome of the CANDU 6 improvement program represents an evolutionary plant design incorporating state of the art features appropriate to 21st century projects, while maintaining provenness via the high degree of design continuity from the existing CANDU 6 plants and current projects.

The enhanced CANDU 6 will incorporate the following key features.

- Advanced control centre.
- Distributed digital control for both safety, process systems.
- CANFLEX fuel.
- Uprated output.
- Improved thermal margins.
- Improved Thermodynamic efficiency via moderator waste heat recycle.
- Simplified containment design and emergency core cooling design.
- Operator grace period of ~ hours.
- Improved equipment status monitoring.
- The use of electronic engineering tools to enhance operator capability to manage plant configuration and plan maintenance.
- Streamlined delivery to achieve short 66 months project schedule.

AECL's development of passive heat sink features will lead to the further stage of CANDU 6 development. This advanced version of the CANDU 6 will include:

- Moderator heat removal using natural circulation for high-reliability accident heat sink.
- Containment heat removal using natural circulation heat exchangers to minimize support system requirements and maximize long-term containment heat removal reliability.
- Optional core design optimized for low-enriched fuel for increased output and fuel burnup.

11. CONCLUSION

This paper outlines the evolutionary future of the CANDU 6 design. A great deal of potential exists to complete the development of the design, generating improvements in plant safety, operability and economics. The result of the AECL design improvements program can be envisaged as an evolutionary, enhanced CANDU 6 design. The next step is already under development, to adapt the design for the inclusion of passive heat sink features, and make available a core design optimized for low-enriched fuel.

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EVOLUTIONARY CANDU 9 PLANT IMPROVEMENTS

IAEA-SM-353/28

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Abstract

The CANDU 9 is a 935 MW(e) nuclear power plant (NPP) based on the multi-unit Darlington and Bruce B designs with additional enhancements from our ongoing engineering and research programs. Added to the advantages of using proven systems and components, CANDU 9 offers improvement features with enhanced safety, improved operability and maintenance including a control centre with advanced man-machine interface, and improved project delivery in both engineering and construction. The CANDU 9 NPP design incorporated safety enhancements through careful attention to emerging licensing and safety issues. The designers assessed, revised and evolved such systems as the moderator, end shield, containment and emergency core cooling (ECC) systems while providing an integrated final design that is more passive and severe-accident-immune. AECL uses a feedback process to incorporate lessons learned from operating plants, from current projects experiences and from the implementation or construction phase of previous projects. Most of the requirements for design improvements are based on a systematic review of current operating CANDU stations in the areas of design and reliability, operability, and maintainability. The CANDU 9 Control Centre provides plant staff with improved operability and maintainability capabilities due to the combination of systematic design with human factors engineering and enhanced operating and diagnostics features. The use of advanced engineering tools and modem construction methods will reduce project implementation risk on project costs and schedules.

1. INTRODUCTION

The evolution of the CANDU® family of heavy water reactors (HWR) is based on a continuous product improvement approach. Proven equipment and systems from operating stations are standardized and used in new products. As a result of the flexibility of the CANDU technology, evolution of the current design will ensure that any new requirements can be met, and there is no need to change the basic concept. CANDU reactors have evolved along two general product lines the CANDU 6 and the CANDU 9. [1]

Building on the success of the 4-unit stations at Bruce B which began commercial operation in 1980s, four additional 900 MW(e) class units were commissioned at Darlington in the early 1990s. The CANDU 9 is a 935 MW(e) reactor based on the multi-unit Darlington and Bruce B designs with some additional enhancements from our ongoing engineering and research programs [2]. Reduced project implementation risk for CANDU 9 has been assured by up-front engineering and licensing prior to contract start.

Added to the advantages of using proven systems and components, CANDU 9 offers improvements providing enhanced safety, a control centre with better operability and a design enabling improved project delivery in both engineering and construction. Enhanced competitiveness of the CANDU product is assured by incorporating improvements based on updated technologies, including safety technology, the rapidly advancing information technology and modern construction methodology.

2. CANDU 9 PROGRAM

The basic engineering work for CANDU 9 followed the product design requirement definition work and conceptual studies, which were started in 1993. The basic engineering program was a 39 month program started in January, 1995 and was concluded at the end of March, 1998. The scope included performing up-front design engineering and the completion of a licensability review of the CANDU 9 NPP by the Atomic Energy Control Board (AECB) to reduce project implementation risk. In 1997 January, the two year licensing review by the AECB was completed. The CANDU 9

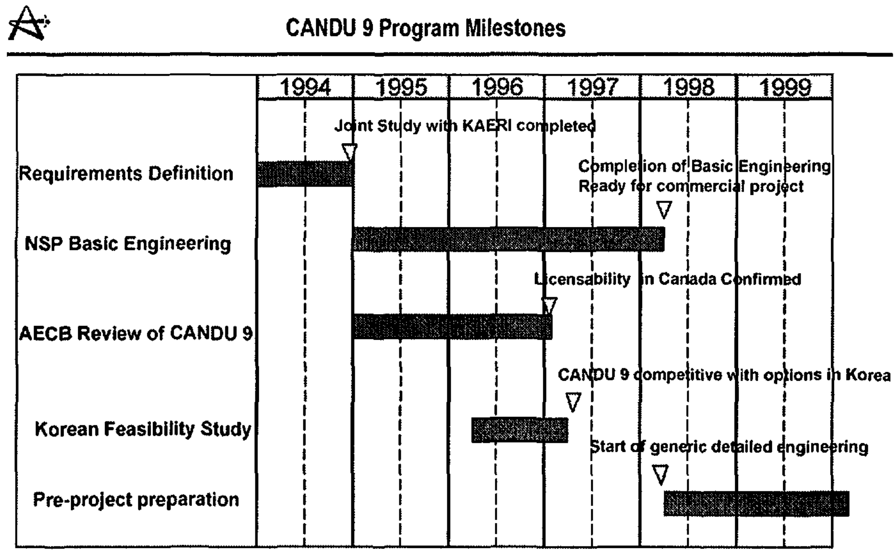

FIG. 1 Major milestones of CANDU 9 program

licensability report by the AECB noted several improvements in the design and confirmed that there are no conceptual barriers to licensing the CANDU 9 in Canada. [3] The overall program milestones are shown in figure 1.

In 1996, a Korean feasibility study comparing CANDU 9 with other NPP options was carried out. AECL has submitted many documents in response to questions from the feasibility study team as well as providing inputs to the technical and economic evaluations. The feasibility study concluded that CANDU 9 is competitive with other reactor options in Korea.

We have identified the generic detailed work to be done as part of the pre-project preparation. This program will complete the CANDU 9 design work that is required to support a shorter project schedule, to evolve and progress the CANDU 9 product and to further enhance the CANDU 9 project delivery.

The preparation of the preliminary safety analysis report is progressing. The licensing interactions with the AECB are continuing in order to resolve most of the additional work items raised in the Statement of Licensability.

3. SAFETY ENHANCEMENTS

In the licensability statement, Atomic Energy Control Board (AECB) staff commented favourably on the key features of the CANDU 9 plant that they considered to be improvements over previous CANDU plants:

- a containment which is simplified by the elimination of the dousing system and has a lower design leakage rate, more reliable isolation, better hydrogen mitigation and
- a reactor coolant system with a larger pressurizer and analysis shows a lower power pulse following a large LOCA
- improved layout and separation between steam & water systems and electrical systems, together with the seismic qualification of the Main Control Room
- better provisions for mitigating severe accidents by use of the reserve water system including improved provisions for steam relief from the shield tank

- significant new features include the replacement of conventional valves in the emergency core cooling system with simple rupture disks and floating-ball isolation valves.

The CANDU 9 NPP design work provided the opportunity to enhance safety, through improvement of safety performance and through careful attention to emerging licensing and safety issues. The designers assessed, revised and evolved such systems as the moderator, end shield, containment and emergency core cooling (ECC) systems while providing an integrated design that is more passive and severe-accident-immune. These safety enhancements are highlighted in the following brief descriptions:

3.1 Containment

Previous CANDU containment system designs included pressure suppression systems utilizing high-flow water sprays and, in multi-unit plants, vacuum buildings. These systems have proven to be very reliable. However, customer requirements have evolved, so that newer CANDU designs now will incorporate the large dry containment concept utilizing a steel liner in a conventional post-tensioned concrete building. While the overall severe accident program ensures a balance between preventative and mitigative measures, the role of containment as a mitigation measure is important.

The CANDU 9 reactor building is a steel-lined, pre-stressed concrete structure which provides biological shielding and an environmental boundary (i.e. a pressure boundary in the unlikely event of a loss-of-coolant-accident (LOCA)). The improved CANDU 9 containment system uses a 'large dry' cylindrical steel lined containment without a dousing system to achieve enhanced containment integrity with increased simplicity. The design leak rate is 0.2 % volume/day at design pressure. Because of the lower design leak rate from containment, the exclusion area radius for the siting of CANDU 9 can be as small as 500 meters, significantly reducing site area requirements for CANDU 9 plants. This is an important advantage in the context of meeting siting requirements and land availability.

The containment free volume is sufficiently large so that no pressure suppression system is required in the short term to maintain the post LOCA peak pressure below the design pressure. The long-term containment atmosphere heat sink is provided by the reactor building air coolers. Judicious layout of equipment inside containment results in large, open volumes, with good potential for natural circulation and no apparent hydrogen traps.

The CANDU 9 containment system automatically closes (i.e. buttons-up) all reactor building penetrations open to the containment atmosphere when an increase in containment pressure or radioactivity level is detected. Automatic isolation of the ventilation lines penetrating the containment structure has been enhanced and is provided by two separate and independent systems for increased reliability. The containment ventilation system provides enhanced atmospheric mixing within the reactor building following a postulated loss-of-coolant-accident. Higher ventilation air flow rates promote better mixing, backed-up by hydrogen igniters/recombiners carefully located at strategic possible accumulation locations. Passive catalytic recombiners are provided to control hydrogen concentration in the long term period after a LOCA; short-term control is accomplished by igniters (the same as those used in current CANDU 6 units).

3.2 Reactor Coolant System

Two improvements were made to the principal process system of the CANDU 9, the Heat Transport System (HTS), relative to the reference design (Bruce B). One improvement consists of interlacing the feeders so that adjacent channels are alternatively connected to separate inlet and outlet headers. In this way the fuel channels served by each inlet header are uniformly distributed throughout the core. This interlaced arrangement minimizes the positive reactivity insertion as a result of a large pipe break in the HTS. The second improvement is the provision of a larger pressurizer capable of accommodating changes in volume of the reactor coolant in the HTS from full power to shutdown condition at 100°C. This enhances the natural circulation capability of the HTS after transients or accidents causing loss of forced flow, (including a steam line break combined with loss of Class IV electrical power,) by ensuring that full reactor coolant inventory in the HTS can be maintained at any time during cooldown.

3.3 Grouping Redundancy and Additional Seismic Qualification

The concept of grouping and separation of safety related systems has been an integral component of CANDU plant designs for many years. This concept provides physical and functional separation of safety related systems to ensure that common cause events do not impair the capability of all systems to perform essential safety functions. Through this concept, the plant can be shut down, decay heat removed, and the plant conditions monitored from systems and components of one of two groups known as Group 1 and Group 2. For the CANDU 9 design, this concept has been enhanced through additional redundancy and diversity in the provision of cooling water and power supplies. In addition, the emphasis has been placed on plant monitoring and management to ensure that the necessary controls needed to monitor and maintain the controlled shutdown and cooldown condition.

A Group 2 feedwater system has been engineered to supply emergency water to the steam generators automatically, for decay heat removal, for approximately 10 hours. This Group 2 feedwater system provides back-up to the diesel driven auxiliary feedwater pump in the event of a loss of normal redundant feedwater pumps in the Group 1 Feedwater Systems. The new system is seismically qualified and is capable of full steam generator pressure operation so as to cope with all the possible operating conditions in the steam generators. The available operator response times, generally 8 hours, have been attained for accident conditions requiring steam generator heat sinks.

In addition, the main control room and the secondary control area are designed to meet the safety grouping and separation design requirements. The main control room remains functional for all design basis accidents, including external events such as an earthquake, whereas the secondary control area is only required for an event such as a major fire or hostile takeover which may require an evacuation of the main control room. Both control locations are qualified to remain functional during design basis and external events, and the necessary structures and systems have been appropriately hardened and qualified.

3.4 Reserve Water System and Severe Accident Mitigation

The CANDU 9 design provides a large reserve water inventory in a torus shaped tank located at high elevation in the reactor building, above the steam generators. The reserve water tank, provides an emergency water supply for several low pressure cooling loads such as the low pressure emergency coolant injection and the backup feedwater supply, as well as providing a make-up source for the shield tank, moderator and heat transport systems.

The reserve water tank is also connected to the normal end shield cooling circuit. During normal reactor operation, the reserve water tank acts as the head tank for the end shield cooling system using circulation pumps. However, in the event of process failures, such as a loss of forced circulation in the end shield cooling circuit or a loss of cooling water to the end shield heat exchanger, the reserve water tank with the large water inventory acts as a passive heat sink. The layout of the equipment and the piping connection between the end shield of the reactor core and the reserve water tank are designed to facilitate enhanced thermosyphoning for heat removal from the end shield. Due to the large heat capacity of the reserve water tank inventory, the rate of heat-up of end shield coolant and the calandria and end shield tubesheet is slow. This increased time interval will ensure fuel channel integrity for a long period before requiring any operator action. An analysis has been carried out confirming the adequacy of the cooling by thermosyphoning after a loss of flow in the end shield system. [4]

CANDU reactors contain large reservoirs of water that are effective in passively removing heat from the core in the event of severe accidents. The fuel channels are surrounded by normally cooled moderator in the calandria vessel and in addition, the moderator in CANDU 9 is surrounded by a shield tank containing light water for biological and thermal shielding. The CANDU 9 shield tank with improved provisions for steam relief, provides a very large inventory of cooling water to contain overheated core components within the calandria shell and extend the time available before accident management measures are required. A further level of passive heat removal for CANDU 9 is achieved by providing gravity-fed inventory to the calandria and shield tank from the reserve water storage tank.

A unique feature of the CANDU design is the capability to prevent fuel channel melting following a loss of coolant accident, even if the emergency coolant injection system does not function. This capability is provided by the cool, low-pressure moderator system surrounding the fuel channels and about 1cm. away from the fuel. In such a severe accident, the fuel would collapse inside the fuel channels, the pressure tubes would either balloon or sag into contact with the calandria tubes, and decay heat would then be transferred to the moderator water. The separate moderator cooling system would remove this heat.

In the event of an even more severe accident in which other failure results in the moderator cooling system not functioning, the moderator water would boil-off slowly. Channels would sag, then collapse progressively to the bottom of the calandria tank. Decay heat would be removed by conduction to the shield tank water. The shield tank with water make-up can absorb decay heat either from the moderator or from debris inside the calandria vessel, and would prevent the core from melting through to containment for tens of hours, until the water had boiled away. The essential safety advantage of this design is the long time taken before fuel might penetrate the calandria tank, allowing mitigation actions to be taken. In the worst case only a small amount of fuel would reach the molten state during this time.

Significant work has been also done for the input to the design of the post-accident management system. Operator Response Guidelines (ORGs) have been prepared for several abnormal events, including LOCA with ECCS available, steam line break, and loss of Class IV electrical power. ORGs for additional abnormal events are under preparation.

3.5 Emergency Core Cooling System

The CANDU 9 emergency core cooling (ECC) system design has been simplified to improve the reliability and performance of the system, to enhance system operation in the event of a Loss of Coolant Accident (LOCA). The key improvements and simplifications made in this special safety system includes the following:

- replacement of D2O isolation valves by one-way rupture discs to separate the heavy water reactor coolant from the light water used for ECC
- elimination of high pressure injection valves
- location of ECC water tanks inside containment
- shorter ECC water injection lines

These simplifications for the ECC has increased the reliability target for this special safety system over that of previous designs. Elimination of the isolation and injection valves avoids all the problems that can be encountered during ECC valve testing due to excessive leakage of the isolation valves and spurious core by-passing due to errors in valve testing sequence at operating stations. These improvements also reduce the capital cost as well as significantly reducing the operating and maintenance costs for testing, inspection, maintenance and repair over the lifetime of the NPP.

3.6 Radiation Protection

The CANDU 9 plant has been designed to comply with ICRP-60, the recommendations of the International Commission on Radiological Protection issued in 1991. Further, as design targets, the CANDU 9 plant has been designed so that total worker exposures will be less than 1 person-Sv/a and the maximum exposure to a member of the public will be less than 50 μSv/a. The approach taken to reduce the internal exposures of workers and tritium emissions to the public has been to reduce tritium-in-air levels by very careful design of the in-containment vapor-recovery system.

To reduce the external exposures of workers during shutdown conditions, the layout has been improved; shielding has been enhanced and corrosion-product activity transport has been reduced. Examples are:

- CANDU 9 carriage maintenance can be carried out in the fuelling machine (F/M) Lock while the reactor is still on power
- Strategically placed shielding on the carriage will allow maintenance to be carried out with reduced dose from the F/M.

Relatively easy access to the reactor building during plant operation always has been a hallmark CANDU advantage. By careful attention to segregation of possible high activity water vapor from areas with lower activity from tritium, this access can be retained while achieving stringent targets for total worker exposure. In addition, fuel handling operations have been reviewed and design features improved to reduce heavy water loss. Examples are:

- New fuel storage has been moved from inside containment to outside the reactor building.
- The fuelling machine (F/M) discharge ports for gravity drain-down of the F/M have been eliminated to reduce potential heavy water loss.
- F/M snout blowdown is to be incorporated to reduce D2O spillage each time a F/M disengages from a channel.
- Vent ports on the transition section between the irradiated fuel transfer mechanism and the fuel transfer port includes a closed loop condenser circuit which will condense and collect D_2O vapour evaporating or boiling from the fuel bundle surface as it is being transferred.

In order to reduce tritium emissions from CANDU 9, a dryer is provided at the reactor building ventilation system air exhaust. Early analysis indicates that there will be at least a three-fold reduction in tritium emissions relative to Wolsong 1 experience.

4. OPERATIONAL AND MAINTENANCE IMPROVEMENTS

AECL uses a feedback process to incorporate lessons learned from operating plants, from current projects experiences and from the implementation or construction phase of previous projects. Most of the requirements for design improvements are based on a systematic review of current operating CANDU stations in the areas of design and reliability, operability, and maintainability. The CANDU 9 Control Centre provides plant staff with improved operability capabilities due to the combination of systematic design with human factors engineering and enhanced operating and diagnostics features.

In the CANDU 9 design, emphasis is placed on improving the reference plant design. The following sections provide some highlights of CANDU 9 plant improvements based on lessons learned from operational experience.

4.1 Reduce Potential Of Process Failures

The CANDU 9 NPP is designed to reduce the frequency of the heat transport system liquid relief valve (LRV) failures and bleed condenser relief valve opening. Its components are specified and designed to ensure adequate relief valve performance in case LRV's fail. The pressure and inventory control system design and the associated procedures identifying necessary operator actions following an LRV failure event minimize the risk of excessive re-pressurization of the bleed condenser which caused the relief valves to open in the Pickering event. Also the CANDU 9 plant reduces reliance on operator action following an LRV opening incident.

The control instrumentation and electrical design in CANDU 9 are coordinated and checked so that logic changes will provide expected response and that failure effects are known. Distribution of the control systems amongst the various partitions, stations and modules within the distributed control system (DCS) is done so that a DCS module failure will not cause multiple system failures and that the scope of the failure is limited and distinct. An independent assessment analysis will be performed to confirm partitioning of control systems implementation on the DCS. As well, an independent hazards impact assessment is conducted to ensure that unexpected fault propagations will not occur.

4.2 Improve Plant Operability

Recent statistics show that high numbers of plant significant events have been directly attributable to human errors. Consequently, special attention has been given to human factors engineering (HFE) during the design of the CANDU 9 nuclear power plant, establishing an HFE design process basis and integrating this HFE process into the project design to interface all designers from all disciplines.

CANDU plants have employed computerized control systems since the 1960s, and each new plant has been provided with then state-of-the-art systems for optimum performance. AECL's strategy for advanced control center design is to extend the proven features of operating CANDU reactors by combining this experience base with operations enhancements and design improvements.

For improved operational capabilities, the CANDU 9 design has incorporated an advanced control centre.[5] The control centre features standard panel human-machine interfaces that provide an integrated display and presentation philosophy; and includes the use of a common plant display system for all consoles and panels. A large, central overview display presents immediate and simplified plant status information to facilitate operation staff awareness of the plant situation in a very legible and recognisable format. A powerful and flexible annunciation system will provide extensive alarm filtering, prioritising and interrogation capabilities to enhance staff recognition of events and plant state.

A major evolutionary change from previous CANDUs is the separation of the Control and Display/Annunciation features formerly provided by the digital control computers. The CANDU 9 plant monitoring, annunciation, and control functions are implemented in two evolutionary systems: the Distributed Control System (DCS) and the Plant Display System (PDS). The DCS implements most of the plant control functions on a single hardware platform while PDS similarly implements the main control room display and annunciation functions. This permits extensive control, display or annunciation enhancements within an open architecture.

An advanced annunciation feature in CANDU 9 is the provision of special safety system impairment levels and potential operating policy violation alarms (i.e. which could occur, for example, in a maintenance mode) with a display/report capability which details the resultant system unavailability or margin encroachment under the prevailing failure or configuration changes. The utilization of a flexible navigation system for the visual display unit (VDU)-based plant display system allows custom information displays to be accessed in a simple, direct, convenient and logical manner by operations or maintenance staff.

The reactor shutdown computers for CANDU 9 include automated system testing and on-line neutronic trip calibration capabilities. One specific benefit of on-line calibration is the provision of an improved "margin-to-trip" thus eliminating unnecessary spurious trips. Safety system monitor computers will provide automated safety system testing, resulting in shorter test duration with reduced opportunity for human error.

A full-scale mockup of the main control centre panels and consoles has been built and is being used for conceptual evaluation, rapid prototyping, design decision-making, and then to allow verification and validation of the interactions between the operator and the annunciation/monitoring/control interface features of the plant. The design of each system of the plant is reviewed against human factors requirements to ensure that the needs of the plant personnel for the monitoring, annunciation and control of the system are appropriately addressed.

4.3 Protection Against Degradation or Ageing Mechanisms

A Plant Life Management program is being undertaken by both the utilities and AECL to safeguard the operating plant investment, and to incorporate the improved knowledge into AECL's latest CANDU 6 and CANDU 9 products.[6] In advance of the formulation of the integrated approach described in the PLIM paper, preliminary results from a number of ageing studies for critical components were incorporated into CANDU 9 to achieve design life as described below.

For the CANDU 9 NPP, feeder material has been specified at a minimum of 0.2 wt% of Chromium content and chemistry control has been specified to achieve a tighter and lower pH operating range. This will reduce concerns with flow accelerated corrosion on the heat transport system side, and particularly in outlet feeders. For CANDU 9, AECL has also adopted strict velocity limits for high energy piping.

An ultrasonic inspection program of the wall thickness in the secondary side piping to monitor thickness variations is recommended as an added assurance against incidents such as feedwater line breaks which occurred in the US. In addition, copper alloy material is avoided in the secondary side; velocity limits are adhered to in the steam and feedwater system design; and alloy steels (whereas 2.25% Chromium and 1% Molybdenum) are used where high erosion is anticipated.

Most piping vibration has been associated with high pressure drops across pressure reduction orifices. Multiplate orifices used in CANDU 9 have been reviewed to eliminate cavitation and to avoid vibration, such as those used in heavy water feed pump and ECC pump by-pass lines, and in the heat transport system balance line.

4.4 Enhance Maintainability and Inspection

The following improvements have been made to CANDU 9 equipment for maintenance and inspection:

4.4.1 Fuel Handling Equipment

- Torlon spacer balls have been added into the CANDU 9 ballscrew assembly to reduce ball wear and increase the number of channel visits between maintenance.
- Use of oil has been reduced in the CANDU 9 system which will minimize leakage and in turn reduce the potential for fire hazard.
- The fuelling machine carriage use more common commercial equipment and sub-assemblies have bolted connections for easy removal and replacement.
- F/H Process equipment layout has been arranged to provide more ready access for maintenance, removal and replacement
- New fuel transfer equipment for CANDU 9 are outside containment which is more readily accessible for maintenance. By transferring the irradiated bundles directly into basket modules, manual bundle handling is considerably reduced later, when preparing for dry storage.

4.4.2 Reactivity Control Units

- Vertical RCU guide tube tensioning springs at the reactivity mechanisms deck facilitate any required re-tensioning and reduces maintenance radiation exposure.

- Horizontal flux detectors will be tensioned by spring and not bellows, this avoid cutting and re-welding bellows for re-tensioning to reduce maintenance radiation exposure.
- Offset LISS nozzles allow more adjustment to avoid contact with sagged calandria tubes due to creep thus eliminating the need for replacement.
- Horizontal RCUs access from outside vault wall will reduce maintenance radiation exposure.

4.4.3 Steam Generators Inspection And Cleaning

Additional inspection ports are added near the tubesheet for CANDU 9 to provide increased access for cleaning, inspection and water lancing. Additional inspection ports were added at each tube support plate on the secondary side to facilitate inspection and cleaning.

5. IMPROVED DELIVERY

CANDU designs utilize advanced engineering tools, such as 3-Dimensional (3-D) Computer Aided Design and Drafting System (CADDS) tools and advanced construction methods, for better economics and reduced risks to future owners. The 3-D CADDS model is used to establish the layout configuration, optimization of the fabrication sequence and construction, and the choice of pre-fabricated structural assemblies depending on the layout and complexities of systems. AECL has developed additional tools to extract component properties directly from the model to carry out necessary analyses. Data are also used to carry out further design detail work such as locating electrical cable runs, specifying pipe hangers as well as conducting stress and seismic analyses.

The computerized engineering tools are modified to access a common project database. For example, the design is progressed using a standardized material database catalogue so that a correctly qualified component is easily specified in a traceable manner for an application. This use of an integrated database will enhance standardization, reduce inventory stocking costs as well as eliminate costly incorrect specifications requiring rework while providing a tool for the utility for on-going configuration management. Parts lists can be taken directly from the model at procurement time for a given project.

Construction, installation, and layout design considerations have resulted in a shorter construction schedule for the CANDU 9 NPP. The ability to reduce the construction schedule was possible due to the adoption of sequence efficient 'open-top' reactor building construction technology utilizing a very heavy lift crane. Other techniques employed to reduce the construction schedule includes parallel fabrication and construction activities, eliminating or reducing construction congestion, providing adequate access and transportation corridors, providing flexible equipment installation sequences as well as reducing material handling requirements.

In addition to these layout improvements for construction improvements, the building layout of the CANDU 9 design results in a narrow 110 meter wide "footprint" that allows several units to be constructed adjacent to each other to form a very compact multi-unit station for better site utilization.

6. SUMMARY

The CANDU 9 is a 935 MW(e) NPP evolved from the multi-unit Darlington and Bruce B designs, with some additional enhancements from ongoing CANDU engineering and research programs. The Basic Engineering program has been successfully completed. AECL has submitted the CANDU 9 design to the Canadian nuclear regulator (AECB) for review, and it was confirmed that there are no conceptual barriers to licensing the CANDU 9 in Canada.

The CANDU 9 NPP engineering work provided the opportunity to enhance safety, through improvement of safety performance and through careful attention to emerging licensing and safety issues. The CANDU 9 designers have made evolutionary improvements to plant safety features matching current requirements.

Feedback from operating experience leads to improvements for improved operability and maintenance including a control centre with advanced man-machine interfaces. Information from PLIM programs underway with operating utilities will assure protection against degradation mechanisms.

The use of advanced engineering tools and modem construction methods will improve project delivery in both engineering and construction, and hence reduce project implementation risk on project costs and schedules.

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ADVANCED PASSIVE PWR AC-600 — DEVELOPMENT ORIENTATION OF NUCLEAR POWER REACTORS IN CHINA FOR THE NEXT CENTURY

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Abstract

Based on Qinshan II Nuclear Power Plant that is designed and constructed by way of self-reliance, China has developed advanced passive PWR AC-600. The design concept of AC-600 not only takes the real situation of China into consideration, but also follows the developing trend of nuclear power in the world. The design of AC-600 has the following technical characteristics: Advanced reactor: 18~24 month fuel cycle, low neutron leakage, low power density of the core, no any penetration in the RPV below the level of the reactor coolant nozzles; Passive safety systems: passive emergency residual heat removal system, passive-active safety injection system, passive containment cooling system and main control room habitability system; System simplified and the number of components reduced; Digital I & C; Modular construction. AC-600 inherits the proven technology China has mastered and used in Qinshan II, and absorbs advanced international design concepts, but it also has a distinctive characteristic of bringing forth new ideas independently. It is suited to Chinese conditions and therefore is expected to become an orientation of nuclear power development by self-reliance in China for the next century.

1. INTRODUCTION

Around the beginning of next century, the development of nuclear power in the world will go into a new period of updating and upgrading. People universally follow with interest which type of advanced new reactors will be adopted in new generation of nuclear power plants. In many years, nuclear power suppliers in the world have been throwing in a great amount of man-power, material and financial resources to research into and develop, in accordance with utility requirements, a new generation of nuclear power reactors which is safer, more economic and more reliable. A lot of positive results of the research and development have been obtained. Among various reactor types that are geared to the needs of next generation of nuclear power plants, a revolutionary type of advanced PWR with passive safety systems has attracted the attention of many people because of its higher level of safety and economy.

On the basis of the nuclear power development policy of "taking ourselves as the dominant factor and co-operating with foreign countries", China has closely been following the trend and progress of international development of nuclear power from the 1980s, and launched research on new technology of next generation of nuclear power based on the national conditions. It is under this background that advanced passive PWR AC-600 has been developed on the basis of Qinshan II Nuclear Power Plant which is designed and constructed by way of self-reliance, AC-600 embodies the features of an advanced reactor: simplified systems, passive safety, digital I&C, and modular construction. Its safety and economy is improved. AC-600 inherits the proven technology China has mastered and used in Qinshan II, and absorbs advanced international design concepts, but it also has a distinctive characteristic of bringing forth new ideas independently. It is suited to Chinese national conditions and therefore is expected to become an orientation of nuclear power development by self-reliance in China for the next century.

Table I. Major Design Targets for AC-600

Parameter	Design Target
Construction cost	About 20% less than that of Qinshan II NPP
Core melt frequency	1×10^{-6}- to 1.5×10^{-6} /r-y
Availability factor	>85%
Refueling period	18 months
Construction period	4~5 years
Plant life time	60 years
Plant personnel exposure dose	0.5-1.0 man-Sv/year

The AC-600 has a large safety margin of operation because of the low power density of the reactor core. The high natural circulation cooling ability due to a small flow resistance of the primary system loop is very useful for reactor core decay heat removal during accidents. The AC-600 has a large reactor pressure vessel, a large pressurizer and a large water volume in the primary systems, which facilitate accident mitigation. The AC-600 design, which eliminates the high-head safety injection pumps, utilizes full pressure core makeup tanks and larger accumulators for the engineered safety features. The passive containment cooling system is used as the ultimate heat sink. All the measures mentioned above increase both the reliability and the capacity of the engineered safety very much, largely improving the safety of AC-600. The major design targets of AC-600 are given in Table 1.

The average linear power density of the AC-600 fuel rod is 13.78 kW/m, much smaller than that of Qinshan- II (16.087 kW/m). The small core power density makes for the reactor to have large thermal safety margins for normal operation and accident conditions.

The AC-600 design uses Gd_2O_3 burnable poison to reduce the excess reactivity of the reactor and the critical boron concentration. Because of the small critical boron concentration, a large negative temperature coefficient of reactivity can be obtained. The small excess reactivity and the large negative temperature coefficient of the core is one of the AC-600 design characteristics, largely improving the passive and inherent safety of the reactor to prevent power excursions induced by reactivity accidents.

The measures of elevating the vertical distance between the steam generators and the reactor core, and reducing the flow resistance, are used in the AC-600 design to increase the natural circulation cooling flow rate of the primary coolant. If the reactor operates at 25% of rated power, the natural circulation flow rate is 4852 t/h=1347.78 kg/s (15.12% of the rated flow rate) after the reactor coolant pumps shut down. The natural circulation flow rate increase is a very important part of the passive safety functions of the AC-600.

The passive emergency residual heat removal system on the secondary circuit side is mainly used in the event of station blackout, main steam line rupture or loss of feedwater supply. The system consists of an emergency feedwater tank, an emergency air cooler, valves and pipes for each loop. When station blackout occurs, the decay heat generated in the reactor core can be removed through use of natural circulation flow in the primary coolant system, in the secondary coolant system, and to the atmosphere, respectively.

In order to increase the reliability of the safety injection system, two full pressure core makeup tanks, two accumulators and four low-head safety injection/recirculation pumps, which are installed in the containment sumps, are utilized in the AC-600 design. In case of a large LOCA, the flow rate into the RCS from a core makeup tank is larger than that from a high-head safety injection pump in the conventional design. It is necessary for the AC-600 to use active pumps to perform the functions of the low-head safety injection/recirculation system.

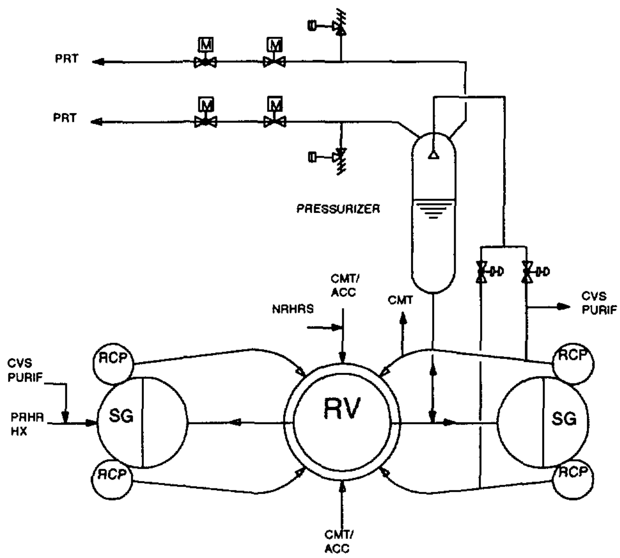


FIG. 1 Reactor coolant system flow diagram

The passive containment cooling system is used to remove the heat from the inside to the outside of the containment during a LOCA or a main steam line rupture inside the containment. First, the water in the tank on the top of the containment will be sprayed out to the outer surface of the steel shell of the containment by gravity, cooling the shell so as to decrease the pressure and the temperature inside the containment. After the tank on the top of the containment becomes empty, the natural circulation flow of air through the annulus between the steel shell and the concrete shell can remove the heat from the inside to the outside of the containment continuously. At the same time, the low-head safety injection/recirculation pumps, which are installed in the containment sumps, can return the borated water from the sumps into the reactor coolant system. The water absorbs the core decay heat and flows out through the break point (in LOCA condition).

2. DESCRIPTION OF THE NUCLEAR STEAM SUPPLY SYSTEMS

2.1 Primary circuit and its main characteristics

The AC-600 reactor plant is based on Qinshan II (2x600 MWe PWR NPP). But it is improved and enhanced in safety, reliability and economy as compared to Qinshan II. The major improvements include:

- utilization of advanced core design with low specific power, gadolinium in fuel, improved fuel management, arrangement of radial reflector and gray control assemblies;
- simplification of reactor coolant system by using high-inertia, high-reliability and low-maintenance hermetically sealed and canned motor pumps connected directly to the bottom of steam generators;
- utilization of passive safety systems to reduce the dependence on power supply;
- introduction of digital I&C systems emphasized on operability and maintainability, reliability and availability, and standardization and modularization;
- application of modular construction technology to save labor and time during construction and maintenance.

The primary circuit of the AC-600 uses two loops - with a steam generator and two reactor coolant pumps in a "one-hot-leg-two-cold-legs" arrangement - connected in parallel and symmetrically to the reactor, a pressurizer, and a relief tank. The schematic flow diagram of the reactor coolant

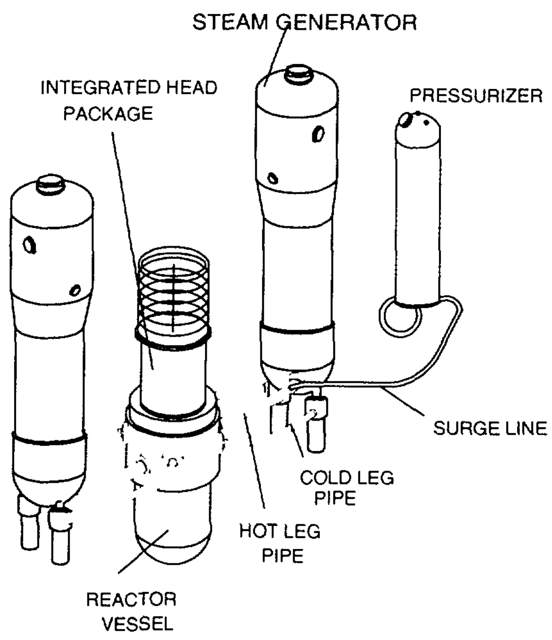


FIG. 2 Isometric view of the reactor coolant loops and major components

system and isometric view of the reactor coolant loops and major components are shown in Figures 1 and 2.

2.2 Reactor core and fuel design

The reactor core consists of 145 17x17 advanced fuel assemblies, 17 control rod assemblies and other fuel associated assemblies. There are 45 black rod (Ag-In-Gd) and 12 gray rod (stainless steel) assemblies in the core. The burnable poison (Gd_2O_3) is dispersed in the fuel. The average burn-up is 42000 MWd/tU.

The control rod drive mechanism (CRDM) for conventional PWR will be adopted in the design for the AC-600, except that wires to be used in the electromagnetic coils for the AC-600 CRDM are melting-extruded. The operating temperature of the coils will be higher than 300 °C (about 350 °C).

2.3 Primary components

2.3.1 Reactor Pressure Vessel

A schematic drawing of the reactor pressure vessel and internals is shown in Figure 3. The reactor vessel encloses all components of the reactor core. It is made of SA 508-3 steel made in China. Because of the lower power density in the core, the energy storage in fuel elements is less and the cladding temperature would be lower under accidental conditions. That results in increased thermal and safety margins. The nozzles of the CRDM and the in-core instrumentation are located on the closure head. There are no penetrations in the reactor pressure vessel below the level of the reactor coolant nozzles.

2.3.1.1 Reactor Internals

The reactor internals consist of two major assemblies—the lower internals and the upper internals. The main function of the reactor internals is to provide protection, alignment and support for the core and control rods to maintain safe and reliable reactor operation. In addition, the reactor internals help to direct the main coolant flow to and from the fuel assemblies, to support instrumentation within the reactor vessel and to provide protection for the reactor vessel against excessive radiation exposure from the core.

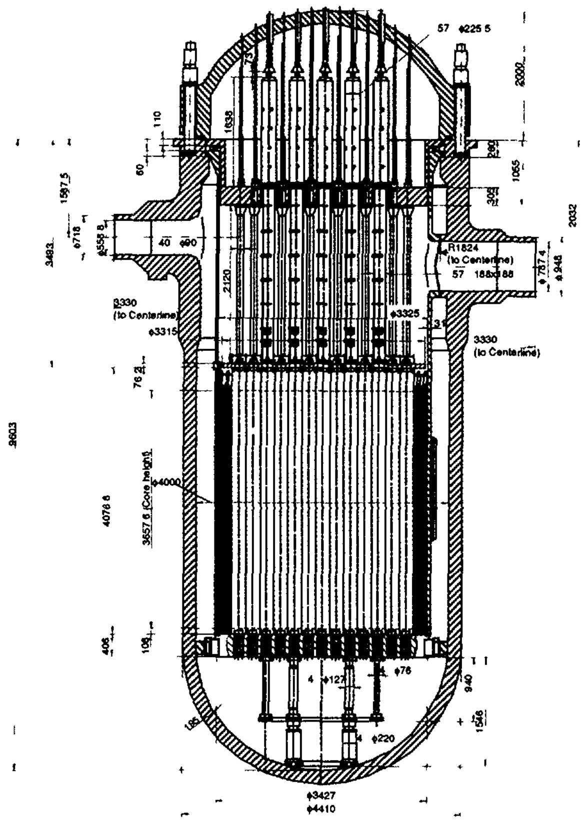


FIG. 3 Reactor pressure vessel and internals

The lower core support assembly consists of the core barrel, lower core support plate, secondary core support, vortex suppression plate, radial reflectors, radial supports, and related attachment hardware.

The upper core support assembly consists of the upper support, the upper core plate, the support columns, and the guide tube assemblies.

2.3.1.2 Steam Generators

The steam generator is of the vertical U-tube type. The material of the U-tubes is thermal-treated Inconel-690 (I-690TT). Two canned-motor pumps are welded to the steam generator bottom head. In this case, the U-shaped cross-over leg of the coolant pipe is eliminated.

2.3.1.3 Pressurizer

The pressurizer for AC-600 is of conventional design. It is about 30% larger than that normally used in a plant of comparable power rating, so as to increase transient operation margins.

2.3.1.4 Reactor Coolant Pumps

The reactor coolant pump (RCP) is of the mixed flow, canned-motor pump type. There are four canned-motor pumps connected to the two steam generator bottom heads directly. Lubrication and cooling of the RCP are performed with water. In order to ensure an adequate inertia of the canned-motor pump, a motor and pump design with a rotating inertia of 0.15 t-m^2 will be employed.

2.3.1.5 Primary Coolant Piping

Reactor coolant system piping is configured with two identical main coolant loops, each of which employs a single 787.4 mm inner diameter hot leg pipe to transport reactor coolant to a steam generator. The two reactor coolant pump suction nozzles are welded directly to the outlet nozzles on the bottom of the steam generator channel head. Two 558.8 mm inner diameter cold leg pipes in each loop (one per pump) transport reactor coolant back to the reactor vessel to complete the circuit.

2.4 Reactor auxiliary systems

The reactor auxiliary systems include chemical and volume control system, equipment cooling system, waste processing and draining system and fuel storage, transfer and handling system.

3. INSTRUMENTATION AND CONTROL SYSTEMS

3.1 Design concepts including control room

The monitoring and control system provides an automated diagnosis of the state and the operating conditions of the NPP. Monitoring and presentation of information on the reactor coolant system, on all the safety-related systems, on the containment, on all operation conditions of the NPP and on remote control of these systems is provided. A post-accident monitoring system is provided to estimate the state of NPP.

Facilities for presentation of information including displays for monitoring of safety systems ensure:

- indication of control rod position
- monitoring of neutron flux during operation, refueling and maintenance
- monitoring of level of radioactive contamination of the containment and the surrounded area
- preservation of adequate water level in the reactor vessel and the cooling systems
- scram (emergency protection) of the reactor
- protection of safety-related systems.

In case of a main control room failure, the reserve control room is to provide:

- reactor trip to hot shutdown condition
- maintaining of hot shutdown condition
- monitoring of sub-criticality
- putting into operation of confining systems
- reactor cooldown with some local operations.

Plant process control systems fulfil the automatic control of the following main controlled parameters:

- neutron flux in the core
- primary pressure
- secondary pressure
- water level in the steam generators
- water level in the pressurizer

The design value of the reactor neutron flux is maintained with the control bank of neutron absorbers, consisting of several rod cluster control assemblies, within ±2% of its nominal value.

The design value of the primary pressure is maintained by the pressurizer electric heaters and by spray valves on the pressurizer spray line from the reactor coolant pump exit side to the steam phase of the pressurizer within ±0.3 MPa.

The design value of the secondary pressure is maintained by an appropriate balance of reactor power and steam flow from the steam generators to the turbine or to the steam dumping devices within ±0.2 MPa.

The water level in the steam generators is maintained within ±180 mm of its nominal level by means of the steam generator feedwater supply controller actuating the control valve on the steam generator feedwater line.

The water level in the pressurizer is maintained within normal value by the level controller, actuating the control valves located on the make-up line, and make-up pumps.

During normal operation, the reactor's neutron power and the process parameters are maintained automatically by the reactor control system. Protection against transients due to the introduction of reactivity is ensured by the reactor protection system. When reaching the set-points of neutron flux or reactor period, the reactor protection system will warn the operators to take actions or will trip the reactor so that reactor safety can be ensured.

3.2 Reactor protection system and other safety systems

The list of automatic safety systems encompasses

- reactor scram (emergency protection) system;
- primary overpressure protection system;
- emergency core cooling system;
- system of passive residual heat removal from the secondary side of the steam generator;
- passive cooling system from the containment;
- system of quick-acting isolation valves in the main steam-lines;
- secondary overpressure protection system;
- diesel-generator system and
- system of reliable direct current power supply.

When any accidents occur, the reactor protection system is designed to perform reactor trip. Scram (emergency protection) is actuated by de-energizing the control rod drive mechanisms.

The following parameters provide input to the reactor protection system:

- increase of neutron flux
- decrease of reactor period
- dT/dt and dP/dt
- decrease of pressure in the reactor
- increase of pressure in the reactor
- decrease of the flow rate in the reactor
- decrease of water level in SG
- increase of water level in SG
- decrease of reactor coolant pump speed
- signal of safety injection.

The reactor protection system trips the reactor, meeting the design criteria under all design conditions.

To mitigate the consequences of ATWS, the protection actions will be taken to trip the turbine and to start the auxiliary feedwater system.

3.2.1.1 Reactor scram (emergency protection) system

The reactor scram (emergency protection) system provides reliable switch-off of the electric power supply to the rod drives, causing the scram (emergency shutdown) rods to drop into the core. In this case, a disappearing signal for the original cause does not stop the initiated scram (emergency protection) action.

3.2.1.2 Primary circuit overpressure protection system

The system comprises three identical pilot safety valve assemblies, which discharge steam or steam-water mixture from the steam phase of the pressurizer to the relief tank when the pressure in the pressurizer increases above the permissible one. The subsystem for receiving the steam or steam-water mixture involves a relief tank and pipelines connecting it with the outlets of the safety valves.

3.2.1.3 Emergency core cooling system

The emergency core cooling system (ECCS) comprises the following complex of subsystems initiated automatically:

- subsystem of core make-up tank with full pressure (high-pressure safety injection subsystem)
- subsystem of accumulator pressurized by nitrogen
- subsystem of low pressure active safety injection and recirculation

Except for the subsystems of low-pressure active safety injection and recirculation, sources of alternating current are not required for the fulfillment of ECCS functions. The air-operated valves, needed for the function of emergency heat removal, are driven by compressed air from the compressed air storage tanks. The power supply of the subsystems of low-pressure active safety injection and recirculation are provided by the diesel generators or by the offsite power source (during the recirculation stage after LOCA).

3.2.1.4 System of passive residual heat removal from the secondary side of the steam generator

The passive residual heat removal system removes the residual reactor power during a plant blackout with the aid of natural circulation on the secondary side of the steam generator (S.G.). It consists of two independent trains, each of them being connected to the respective reactor loop via the S.G. Each train has an emergency feedwater tank, a heat exchanger cooled by air and located outside the containment, and piping for steam and condensate circulation. The fail-open valves on the piping are driven by compressed air. The air-cooled heat exchanger rejects decay heat via the steam generators into the atmosphere outside the containment.

3.2.1.5 Passive cooling system from the containment

The passive containment cooling system removes heat from the containment in the event of loss-of-coolant-accident (LOCA) in the primary circuit. Steam released will condense on the inside of the containment shell that is cooled on the outside by natural circulating air and gravity drain of water from elevated tanks above the containment. The heat released to the inside of the containment is rejected to the atmosphere from the outer surface of the containment. The pressure of the atmosphere inside the containment is kept below the permissible design value.

3.2.1.6 System of quick-acting isolation valves in steam lines

The quick-acting isolation valves in the steam lines close at:

- water inventory in steam generators increases above the permissible level;
- increase of radioactivity in steam generators above the permissible level; and
- received signals of a steam line rupture.

The system provides for

- protection of the turbine from steam of high humidity;
- prevention of radioactivity release from steam generators; and
- restriction of steam blow-down during rupture of the secondary circuit.

3.2.1.7 Secondary circuit overpressure protection system

This system prevents the secondary circuit pressure to increase above the permissible level of 110% of secondary design pressure. It incorporates a power-operated relief valve and seven safety valves. These valves reject steam into the atmosphere.

4. ELECTRICAL SYSTEMS

4.1 Operational power supply systems

The normal and the emergency electric power supply system consists of two trains of 100% capacity, with each channel being divided into three groups considering reliability requirements and the time interval of loss of electric power.

4.2 Safety-related systems

4.2.1 Diesel generator system

Two physically separated diesel generators provide power supply to the safety-related systems, involving the recirculation pumps of the subsystem of low-pressure active recirculation.

Start-up of the two diesel-generators, one for each channel of reliable electric power and to be put into operation in the case of failure of main and reserve grid connections, is carried out in a time not exceeding 15 s from the moment of generation of a command to start-up.

4.2.2 System of reliable direct current power supply

This system consists of storage batteries. It provides the power supply to electromagnetic circuits for operating of safety systems and for recording of necessary post-accident parameters.

The D.C. electric power supply of the reactor control and protection system is ensured by batteries (in each train) designed for a discharge over 24 hours. Electric power supply from accumulator batteries during a station blackout situation is provided for the main control room and the auxiliary control room in full measure.

5. SAFETY CONCEPT

5.1 Safety requirements and design philosophy

The safety goals of nuclear power plants should include not only the protection of the environment and the public, but the protection of the plants themselves as well. The two sides of the safety goals can not be separated completely but are closely related to each other. It is quite evident that only under the prerequisite of the safety of the nuclear power plants themselves the goals of the environmental safety and the public health can really be achieved. Increasing the plant's own safety and preventing core melt should be emphasized so as to restore the public confidence in nuclear power.

Any intervention of operators during design basis accidents is prohibited for the first 30 minutes so that operators have enough time to consider the features of the accident occurred, which may prevent erroneous actions. Reactor safety actions are performed fully by the automatic control and protection system for the first 30 minutes.

In addition, provisions under design basis accidents are as follows:

- accident state monitoring such as in-core and sump level monitoring
- indication of control rod position, including lights and digits
- indication of radiation level and radioactive releases
- monitoring of the reactor safety shutdown states

The above systems are provided with automatic recording devices during any accidents. Alarm light signals and digital indications are also provided in the main control room.

5.1.1 Deterministic design basis

In the AC-600 design, the safety design basis depends on the Engineered Safety Features. Through deterministic safety studies, some key parameters should be optimized, such as the size of the CMTs, the size and the set-point of the cold leg accumulators, the size of the air coolers and the chimneys on the secondary side of the S.G.s, and the set-points for the control and protection system.

5.1.2 Risk reduction

PRA has played an important role in the AC-600 design. It was used to prove the design for features that could be improved from a risk standpoint. A number of plant design enhancements were made as a result of these PRA studies, and these have yielded a predicted core damage frequency of less than 10E-5/y and a significant release frequency of less than 10E-6/y, including:

- Passive safety-related systems reduce the dependence of safety-related system operation on electric power and compressed air. This significantly reduces the core damage frequency resulting from a loss of offsite power or station blackout event.
- The use of canned motor reactor coolant pumps eliminates pump seal loss of coolant accidents.
- Simplified passive safety-related systems reduce the need for, and importance of, operator action.
- There is no core damage sequence in which high-pressure failure of the reactor coolant system pressure boundary occurs.
- The analysis shows that many of the events, which, in the past, were leading contributors to the risk of nuclear power plants, are not as significant for the AC-600. Interfacing systems loss of coolant accidents, which are typically the highest risk for severe accident sequences, are virtually eliminated by the design of the AC-600.
- There is no containment failure due to hydrogen burns or other energetic phenomena. This is true even in events without water cooling on the outer surface of the containment shell.
- The integrated severe accident containment analysis shows that the AC-600 containment is capable of performing its function as the ultimate fission product barrier, and no containment failures occur from containment over-pressure or over-temperature. Thus, large release of steam

would result only from sequences involving containment bypass or failure of containment isolation.

- The frequency of significant release calculated in the PRA study provides a basis for eliminating the needs for an evacuation plan.

5.1.3 External and internal hazards

In the safety analysis, both external and internal hazards are taken into account. External hazards include natural phenomena such as earthquakes, wind, tornadoes, floods, and external missiles. Internal events are divided into the following five primary categories:

- manual shutdown,
- transients,
- loss of offsite power,
- loss of coolant accidents, and
- anticipated transients without scram.

5.2 Safety systems and features (active, passive, and inherent)

In the first stage of normal plant shutdown, the residual heat of the reactor and the coolant system is transferred to the secondary loop through the steam generators. The steam generated then enters the condenser through the turbine bypass system to be condensed. The auxiliary feedwater system supplies the steam generators with water. The whole process goes on till the pressure of the coolant system drops to 2.8 MPa and the temperature to 180 °C.

In the second stage of shutdown, residual heat removal is accomplished by the residual heat removal system. The residual heat removal system and the spent fuel pool cooling and purification system share the same equipment. It consists of two independent trains, each of which includes one pump and heat exchanger cooled by equipment cooling water. During normal plant operation, this system acts as the spent fuel pool cooling and purification system. During plant shutdown, one of its trains is used to transfer reactor residual heat. At the same time, the spent fuel pool is also cooled till the coolant pressure is under 0.1 MPa. Coolant temperature drops and will be kept at cold shutdown temperature.

During the plant shutdown and cooling process, the coolant pump is always in operation. It does not stop until the coolant temperature drops to 70 °C. Before it stops totally, coolant circulates in coolant loop. After that, coolant is driven by the spent fuel pit cooling pump.

Under the condition of a large leakage from the reactor coolant pressure boundary, emergency residual heat is removed by the emergency core cooling system and the passive containment cooling system. The water volume of the primary system is guaranteed by the emergency core cooling system in order to keep the fuel assemblies in the pressure vessel covered with water.

Under the condition of completely intact pressure boundary, the typical situation in which emergency residual heat removal is required is blackout. At this time, the passive heat removal system on the secondary side of the steam generators is automatically put into operation. Through natural circulation of primary coolant, natural circulation of secondary loop steam and condensed water, and natural convection of air in special ducts outside the containment, residual heat is removed to atmosphere. By this system the coolant temperature and pressure can be brought to the corresponding values for cold shutdown, or till the power supply is restored. Besides the condensed water from the air-cooled heat exchanger, secondary side system feedwater is also available from the emergency feedwater tank. So, the water volume is kept at the required value by natural circulation in the secondary system.

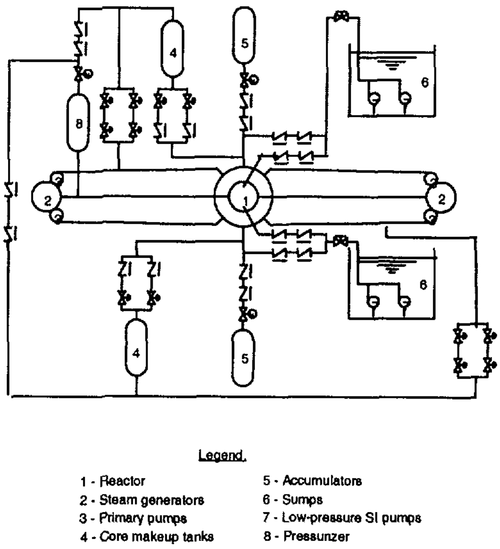


FIG. 4 Passive Safety Injection (PSI) system

5.2.1 Emergency core cooling system

The AC-600 utilizes an emergency core cooling system that is based on the principle of combining passive and active features. There are three subsystems for the emergency core cooling system. The AC-600 passive safety injection system is schematically shown in Figure 4.

The high-pressure injection subsystem consists of two reactor core makeup tanks. The middle pressure injection subsystem consists of two accumulators. The low-pressure injection and long term cooling subsystem consists of four low-pressure injection pumps taking suction from two special sumps in the containment. The low-pressure injection pump is of the vertical, low suction head type. The main functions of the emergency core cooling system are as follows:

- To supply water to the reactor in the event of abnormal leakage,
- In the event of LOCA, to inject water into the reactor and provide long term core cooling.

5.2.2 Passive residual heat removal system

The AC-600 passive residual heat removal system is illustrated in Figure 5. The function of this system is to remove the reactor core residual heat when the reactor loses its normal cooling resulting from station blackout or other accidents.

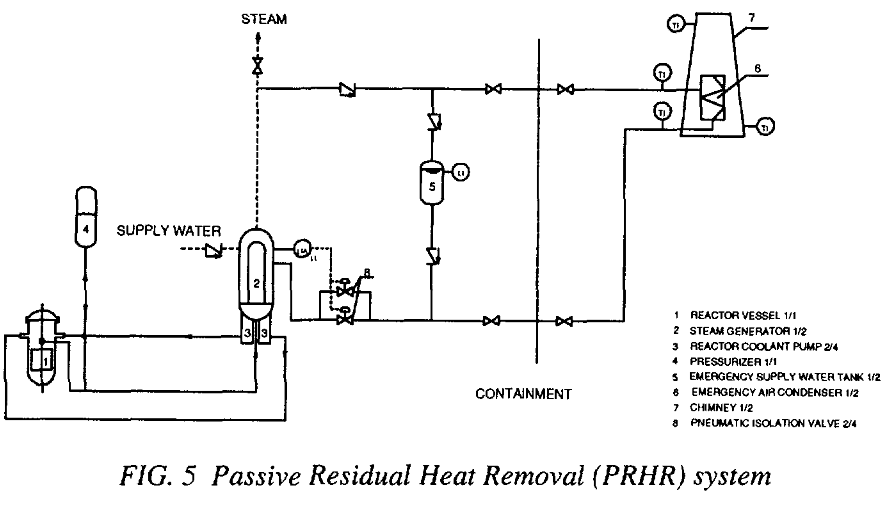


FIG. 5 Passive Residual Heat Removal (PRHR) system

This system has two trains. Each train consists of an emergency water tank and an emergency air-cooler. When blackout or other accidents occur, the isolation valves located at the outlet pipe of the emergency water tank are opened by a low-low water level signal for the steam generator, so that the emergency water tanks provide water to the secondary side of the steam generator by gravity and maintain the water level. The water in the steam generator absorbs the heat from the reactor coolant, when the water is heated into steam. The steam rises and passes through the emergency air-cooler where the steam is condensed into water. Simultaneously, the heat is transferred to the atmosphere. The condensed water returns to the steam generators by gravity, thereby a continuous natural circulation path will be established. Because of the cooling of the secondary side of the steam generators, a corresponding natural circulation in the reactor coolant system will also be established. In this way, the residual heat of the reactor core will be transferred to the atmosphere.

5.3 Severe accidents (Beyond design basis accidents)

5.3.1 Severe Accident Mitigation Strategy

The strategy applied to the AC-600 design to meet the severe accident challenge is a "defense-in-depth" strategy. The strategy includes the following steps:

- Planning accident management (AM) to prevent and mitigate severe accident consequences
- Prevention of high pressure core melt scenarios
- Protection of reactor pressure vessel by means of flooding of reactor cavity
- Protection of the containment integrity by means of control of hydrogen and control of pressure and temperature in containment
- Retention of radioactive material inside the containment and limitation of source terms

5.3.2 Severe Accident Prevention and Mitigation Features

5.3.2.1 Severe Accident Prevention Features

Following the accident initiator, the most important thing is to ensure inventory control and sufficient heat removal from the core. Failure to provide heat removal or inventory control results in core uncovery, fuel overheating and the potential for oxidation and melting of the reactor core.

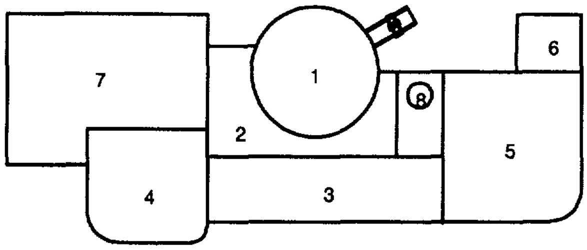
In response to accident initiators identified through operating reactor experience and performance of probabilistic risk assessments, design features for the AC-600 to prevent the occurrence of such initiators from leading to a severe accident are still under study. These initiators include ATWS, mid-loop operation, station blackout, fire and intersystem loss-of-coolant accident.

5.3.2.2 Severe Accident Mitigation Features

External reactor vessel cooling is used to prevent reactor vessel failure and subsequent relocation of melted core debris into the containment. The engineered design features of the AC-600 promote flooding of the reactor cavity and submergence of the reactor vessel lower head.

Generation and combustion of large quantities of hydrogen can threaten containment integrity. So, in-vessel and ex-vessel hydrogen generation must be considered. The AC-600 is equipped with a hydrogen ignition system composed of thermal igniters. The intent of the system is to ignite the hydrogen as soon as sufficient hydrogen has accumulated to achieve a combustible mixture for detonation prevention.

One important boundary condition for severe accident control is that core-concrete interaction (CCI) with its potential for basemat penetration is prevented and hydrogen generation is limited. Many features in the AC-600 design to help mitigate the effects of CCI are still under development.



LEGEND.

1 Containment/Shield Building
2 Auxiliary Building
3 Annex Building
4 Access Control
5 Radwaste Building
6 Diesel Generator Building
7 Turbine Building
8 Secondary Passive Cooling Stack

FIG. 6 General plant layout arrangement

High-pressure core melt ejection (HPME) and subsequent direct containment heating (DCI) can lead to early containment failure with large radioactive release to the environment. The AC-600 design provides a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment.

Steam explosion resulting from fuel-coolant interaction is energetic and the rapid energy release can challenge the containment function. Steam explosion can occur either inside or outside the reactor vessel. On the basis of present knowledge, steam explosion has only a very low probability of occurrence. We expect that further work will confirm this. In any case, the risk of a steam explosion could be minimized by technical means.

The pressure and temperature in the containment can be controlled by means of the passive containment cooling system and filtered venting provided by the AC-600 design.

The equipment needed to perform mitigative function and the environmental conditions under which the equipment must function are under consideration to assure the equipment's survivability in severe accident.

In order to prevent the core from melting or radioactive release from the plant to the environment, operators are required to utilize all reasonable measures according to procedure H or U.

The fission chain reaction in the core should be stopped during severe accidents and the reactor returned to a controllable state. The measures of mitigation and accident management are researched through use of severe accident analysis.

6. PLANT LAYOUT

6.1 Buildings and structures, including plot plan

The plant layout is composed of seven principal building structures: the nuclear island, the turbine building, the annex building, the access control building, the secondary passive cooling stack, the diesel building and the radwaste building, as shown in Figure 6.

The nuclear island consists of a free-standing steel containment building, a concrete shield building, and an auxiliary building. The foundation for the nuclear island is an integral basemat, which supports these buildings. All safety-related equipment designed to perform accident mitigation functions is located in the nuclear island. The containment building houses the reactor coolant system and other related systems. The auxiliary building contains the main control room, instrumentation and control systems, class 1E direct current system, fuel handing area, mechanical equipment areas, containment penetration areas and main steam and feedwater isolation valve compartment.

The turbine building houses the main turbine, generator and associated fluid and electrical systems.

The annex building includes the health physics facilities, the non-1E AC and DC electric power systems, the lunch and cooker rooms, the technical support center and various heat, ventilation and air conditioning systems, the hot machine shop and access way to the upper and lower equipment hatches of the containment building for personnel access during outages.

The access control building includes the health physics facilities for the control of entry to and exit from the radiological control area, access ways for personnel and equipment to the clean areas of the nuclear island in the auxiliary building and to the radiological control area via the annex building.

The secondary passive cooling stack houses two air coolers. It is located on the top of the fuel handling area.

The diesel building houses two diesel generators and their associated heating, ventilation and air conditioning equipment.

The radwaste building contains facilities for the handling and storage of plant waste.

6.2 Reactor building

The reactor building consists of containment building and shielding building.

The containment building is a free-standing cylindrical steel containment vessel with elliptical upper and lower heads, it houses the reactor coolant system and other related systems.

The shield building surrounds the containment building. It is a reinforced concrete structure, with passive containment cooling system air baffle, passive water storage tank, and passive air diffuser. The shield building provides the protection for the containment building from external events and the required shielding for the internal postulated design basis accidents, it is also an integral part of the passive containment cooling system.

6.3 Containment

The containment is a free-standing cylindrical steel containment vessel with elliptical upper and lower heads. It houses the reactor coolant system and other related systems. The containment vessel includes the shell, hoop stiffener and crane girder, equipment hatches, personnel airlocks, penetration assemblies, and miscellaneous appurtenances and attachment. There are two floor elevations (grade access maintenance floor and operating deck) and a few lower equipment compartments within the containment building. Floor gratings are provided for access to equipment at other elevations. The passive core cooling system is located in the containment building. A seismically analyzed polar crane is provided in the containment and its bridge is sized for lifting the steam generator. The polar crane support is attached to the steel cylindrical shell of the containment. Two containment re-circulation heating, ventilation and air conditioning modules are provided. The chemical and volume control system equipment is located in the containment below the grade access maintenance floor level, a refueling machine and a fuel transfer system are designed in the containment.

The containment building provides shielding for the reactor core and the reactor coolant system during normal operations and provides a high degree of leak-tightness; it also is an integral part of the passive containment cooling system. The containment vessel and the passive containment cooling system are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents.

6.4 Turbine building

The turbine building for AC-600 houses the main turbine, generator, condenser, feedwater pump, deaerator and water storage tank, heater and associated fluid and electrical systems. It provides weather protection for the lay-down and maintenance of major turbine/generator components. The turbine building also houses the makeup purification system. No safety-related equipment is located in the turbine building. The turbine building consists of the main house, deaerator room and auxiliary area for condensate polishing.

The turbine building is 100 m long and 57 m wide. A bridge crane is provided for turbine maintenance.

The turbine building is a steel column and beam structure. The turbine building ground floor is a reinforced concrete slab. The foundation for the entire building is a reinforced concrete mat.

6.5 Service buildings

The service building for AC-600 include the protection auxiliary buildings, the buildings in the front area of the plant, the buildings in the depository area, the buildings adjacent to the site, the environment monitoring works, and BOP works.

7. TECHNICAL DATA

7.1 General plant data

Power plant output, gross	636	MWe
Power plant output, net	600	MWe
Reactor thermal output	1930	MWt
Power plant efficiency, net	31.1	%
Cooling water temperature	18	°C

7.2 Nuclear steam supply system

Number of coolant loops	2	
Primary circuit volume, including pressurizer	219	m^3
Steam flow rate at nominal conditions [1951 t/h]	2168	kg/s
Steam temperature/pressure	282.94/6.65	°C/MPa
Feedwater temperature		°C

7.3 Reactor coolant system

Primary coolant flow rate [13.49 m^3/s]	10002	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	292.8	°C
Coolant outlet temperature, at RPV outlet	327.2	°C
Mean temperature rise across core	36.8	°C

7.4 Reactor core

Active core height	3.658	m
Equivalent core diameter	2.922	m
Heat transfer surface in the core	6222.7	m^2
Total fuel weight (UO_2)	66.8	t
Average linear heat rate	13.78	kW/m
Average fuel power density	32.9	kW/kg U
Average core power density (volumetric)	78.69	kW/l
Thermal heat flux, Fq	311.1	kW/m^2
Fuel material	Sintered UO_2	
Fuel assembly total length	4100	mm
Rod array	Square, 17×17-25 (AFA)	
Number of fuel assemblies	145	
Number of fuel rods/assembly	264	
Number of guide tubes for control rods/instr.	24/1	
Number of spacers	8	
Enrichment (range) of first core	2.0, 2.5, 3.0	Wt%
Enrichment of reload fuel at equilibrium core	3.6	Wt%
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel	42000	MWd/t
Cladding tube material	Zr-4	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly	663.2	kg
Active length of fuel rods	3658	mm
Burnable absorber, strategy/material	Gd_2O_3,	Mixed with fuel
Number of control rod assemblies	57 (45 black & 12 gray rods)	
Absorber rods per control assembly	20	
Absorber material , black/gray rods	Ag-In-Cd/stainless steel	
Drive mechanism	Magnetic jack	
Positioning rate	72	Steps/min
Soluble neutron absorber	Boron acid	

7.5 Reactor pressure vessel

Cylindrical shell inner diameter	4000	mm
Wall thickness of cylindrical shell	205	mm
Total height	12220	mm
Shell and head material	Low alloy steel A508-III	
Design pressure/temperature	17.2/343	MPa/°C
Transport weight (lower part)[incl. Head]	440	t

7.6 Steam generator

Type	Vertical, U-tube heat exchanger	
Number	2	
Heat transfer surface area	5631	m^2
Number of U-type tubes	4640	
Tube dimensions	19.05×.1.09	mm
Maximum outer diameter	4843	mm
Total height	21000	mm
Transport weight	~350	t
Shell and tube sheet material	A508-III	
Tube material	I-690TT	

7.7 Reactor coolant pump

Type	Single-stage centrifugal pump with canned motor	
Number	4	
Design pressure/temperature	17.2/343	MPa/ °C
Design flow rate (at operating conditions) [3.374m^3/s]	2500	kg/s
Pump head	72	m
Power demand at coupling, cold/hot	3340/2545	kW
Pump casing material	Stainless steel	
Pump speed	1488	rpm

7.8 Pressurizer

Total volume	36	m^3
Steam volume: full power/zero power	14.4/23.6	m^3
Design pressure/temperature	17.2/360	MPa/ °C
Heating power of the heater rods	1440	kW
Number of heater rods	60	
Inner diameter	2100	mm
Total height	11760	mm
Material	A508-III	
Transport weight	~95	t

7.9 Pressuriser relief tank

Total volume	37	m^3
Design pressure/temperature	0.7/170	MPa/ °C
Inner diameter(vessel)	3000	mm
Total height	5270	mm
Material	Austenitic stainless steel	
Transport weight	~12	t

7.10 Primary containment

Type	Dry, double wall, in steel/concrete	
Overall form (spherical/cyl.)	Cylindrical	
Dimensions (diameter/height)	37/57	m
Free volume	41500	m^3
Design pressure/temperature (DBEs)	430/130	kPa/°C
Design leakage rate	<0.25	vol%/day
Is secondary containment provided?	Yes, space between the walls	

8. FOUNDATIONS AND CONDITIONS FOR AC-600 DEVELOPMENT IN CHINA

8.1 Experimental and research base for nuclear power development

In order to fulfil self-reliance in nuclear power design, the Chinese government devotes much attention to research and development, and has thrown in large funds to construct a set of large test installations and research facilities at the Nuclear Power Institute of China (NPIC), which is located in Chengdu, Sichuam province of China. These installations and facilities are necessary for the AC-600 development. Part of them are listed below.

(1) Test facility for AC-600 passive emergency residual heat removal through secondary side of SG.

(2) AC-600 full pressure core makeup water test facility.

(3) Overall reactor core hydraulic simulation test facility.

(4) High neutron flux engineering test reactor.

(5) Comprehensive test facility of power device (a large loop and a small loop).

(6) Hydraulic test facility for control rod drive alignment under cold conditions.

(7) Comprehensive behavior test facility under hot condition for control rod drive alignment.

(8) Water chemistry test facility.

(9) Large scale thermal test facility.

(10) Well (shaft) used for seismic test and multi-excitation test facility.

8.2 Feedback of Qinshan II engineering experience

AC-600 is developed on the basis of Qinshan II, and its major parameters and main equipment are basically the same as those of Qinshan II, and therefore, a large amount of test and research result for Qinshan II can be used in the development and research of AC-600. The following summarizes some of the tests and research that have been performed for Qinshan II.

(1) Single and overall tests of reactor hydraulic simulations

(2) Control rod drive mechanism (CRDM) test

(3) Reactor protection system development and test

(4) Flow-induced vibration test for reactor internals

(5) Irradiation test of 4×4 fuel assembly in the high flux test reactor

(6) Test and research for reactor control and measurement systems, rod control system, and rod position indicating system

(7) Seismic test for control rod drive alignment

8.3 AC-600 R&D status

Starting from the eighth five-year plan, China has formally brought AC-600 development into line with the key state scientific research plan, and has performed a large amount of work on test and research, and made a lot of progress and breakthroughs. Table 2 lists major test and research items for AC-600 that have been completed. Through these tests and research, the passive safety concept of AC-600 has been verified in principle. The results obtained from these tests confirm the AC-600 design. All of this lays a good foundation for the future work.

TABLE 2. THE COMPLETED EXPERIMENTAL AND RESEARCH ITEMS OF AC-600

Designation	Contents	Finish time
Critical heat flux (CHF) test at low flow rate	The coolant flow per unit area of AC-600 core is relatively low. Under this low-flow-rate condition, measuring test data of departure from nucleate boiling on element surface and draw up a formula.	In 1993
Make-up test for make-up tanks at full pressure	Research on the passive characteristics of the make-up by a makeup tank at full pressure.	In 1997
Wind tunnel test for passive containment cooling system	Research on the correlation between flow resistance, flow duct shape and flow rate of the passive containment cooling system, and research on natural convection cooling characteristics.	In 1994
Test for the passive emergency core residual heat removal system on the secondary side	Research on the capability of the emergency core residual heat removal system on the secondary side, on the flow characteristic of the natural circulation and on the supporting means	In 1995
Flow characteristics test of the baffle plate in the exit plenum of SG	Research on the flow characteristics of the baffle plate in the exit plenum of the AC-600 SG and the entrance plenum of the main pump	In 1995
Digital I&C systems	Research on the control, adjustment and in-core measurement (Pneumatically-driven balls measurement)	In 1995

China is now working out future test and research plans based on what has already been achieved. It mainly includes the following items:

(1) New computer code development in order to be able to make calculations for Gd_2O_3 burnable poison and design for low neutron leakage. Performing zero-power physical test for the Ferro-water reflector in order to verify and improve computer code.

(2) Further improving reactor thermal-hydraulic design, and methodology for accident analysis and protection set-point determination.

(3) Developing computer codes used for passive safety system analysis.

(4) Carrying out digital I&C equipment development stage by stage.

(5) The general test scheme of passive containment cooling is now under proving. And the test facility is under preliminary design.

(6) Comprehensive test and research on passive core residual heat removal system.

(7) Further research on behavior of high performance fuel element and burnable poison rod.

9. CONCLUSION

The AC-600 is developed on the basis of the proven technology that China has mastered and used in Qinshan II NPP. Its design has fully absorbed ideas of advanced design concepts in the world and has a distinctive characteristic of bringing forth new ideas. The safety and economy have been improved. Technically, AC-600 is an advanced design that meets the requirements of nuclear power for the next century. It is suited to the real conditions of China. The self-reliance of nuclear power in China can be realized by means of the AC-600 development. Therefore, AC-600 represents the orientation of self-reliant nuclear power development in China for the next century.

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THE KEY DESIGN FEATURES OF THE INDIAN ADVANCED HEAVY WATER REACTOR

XA0053562

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Abstract

The 235 MWe Indian Advanced Heavy Water Reactor (AHWR) is a vertical, pressure tube type, boiling light water cooled reactor. The three key specific features of design of the AHWR, having a large impact on its viability, safety and economics, relate to its reactor physics, coolant channel, and passive safety features. The reactor physics design is tuned for maximising use of thorium based fuel, and achieving a slightly negative void coefficient of reactivity. The fulfilment of these requirements has been possible through use of PuO_2-ThO_2 MOX, and ThO_2-$U^{233}O_2$ MOX in different pins of the same fuel cluster, and use of a heterogeneous moderator consisting of pyrolytic carbon and heavy water in 80%-20% volume ratio. The coolant channels of AHWR are designed for easy replaceability of pressure tubes, during normal maintenance shutdowns. The removal of pressure tube along with bottom end-fitting, using rolled joint detachment technology, can be done in AHWR coolant channels without disturbing the top end-fitting, tail pipe and feeder connections, and all other appendages of the coolant channel. The AHWR incorporates several passive safety features. These include core heat removal through natural circulation, direct injection of Emergency Core Coolant System (ECCS) water in fuel, passive systems for containment cooling and isolation, and availability of a large inventory of borated water in overhead Gravity Driven Water Pool (GDWP) to facilitate sustenance of core decay heat removal, ECCS injection, and containment cooling for three days without invoking any active systems or operator action. Incorporation of these features has been done together with considerable design simplifications, and elimination of several reactor grade equipment. A rigorous evaluation of feasibility of AHWR design concept has been completed.. The economy enhancing aspects of its key design features are expected to compensate for relative complexity of the thorium fuel cycle activities required to support the operation of this reactor.

1. INTRODUCTION

The Indian Advanced Heavy Water Reactor (AHWR) is being designed as a vertical, pressure tube type, boiling light water cooled and heavy water moderated reactor. The reactor is designed to produce most of its power from thorium, aided by a small input of plutonium based fuel. The reactor will have a large number of advanced safety features, such as passive safety systems not requiring either external power or operator action for activation.

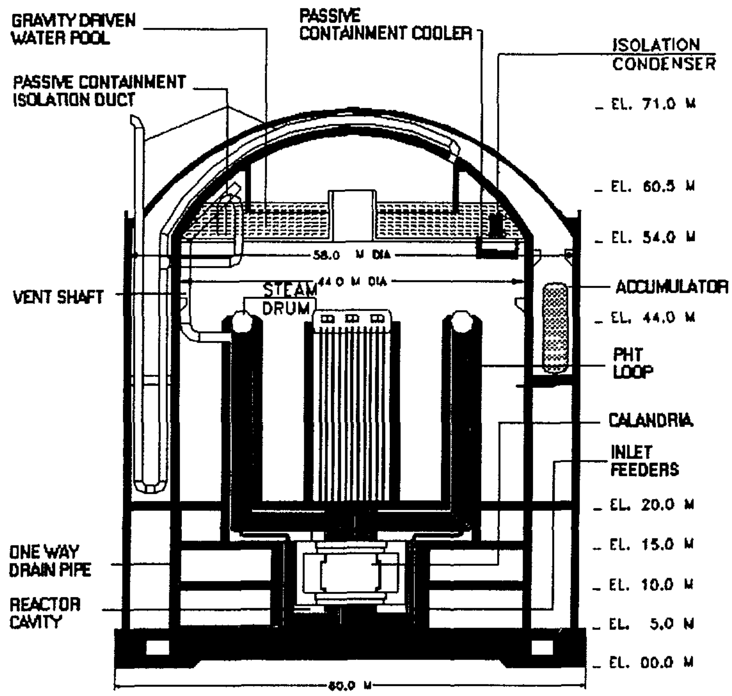
The basic design features of this reactor are indicated in Figure 1. At the current stage of development, the feasibility study for the reactor has been completed. Detailed design of the systems and components is in progress. It is envisaged that the first unit of AHWR will be commissioned in the early part of the second decade of the next century.

The three key features of design of the AHWR, having a large impact on its viability, safety and economics are:

a) Reactor physics design tuned for using thorium based fuel, with negative void coefficient of reactivity

b) Advanced coolant channel design features, with easily replaceable pressure tubes

c) Passive systems for core heat removal, containment cooling and containment isolation



- FUEL :U-233/THORIUM MOX + Pu-239/U-238 MOX
- COOLANT: BOILING LIGHT WATER
- MODERATOR: HEAVY WATER
- POWER: 235 MW(e)
- NEGATIVE VOID COEFFICIENT OF REACTIVITY
- PUMPLESS PRIMARY CIRCULATION
- PASSIVE DECAY HEAT REMOVAL
- ADVANCED EMERGENCY CORE COOLING
- HEAT SINK ALLOWING 3 DAYS GRACE PERIOD
- PASSIVE CONTAINMENT COOLING
- PASSIVE CONTAINMENT ISOLATION

FIG. 1. Advanced Heavy Water Reactor

2. REACTOR PHYSICS DESIGN

2.1 Main Features

The reactor physics design of AHWR has been based on the following main criteria:

a) Reactor power: 750 MWt

b) Fraction of power to be generated in thorium: 75 percent

c) Plutonium inventory in the core: 300 kg, maximum.

d) Slightly negative void coefficient of reactivity:

e) Boiling light water cooled vertical pressure tube type design

f) Burn up of fuel: 20,000 MWD/Te, minimum goal for prototype.

The fulfilment of these requirements has been possible through the use of PuO_2-ThO_2 (MOX), and ThO_2-$^{233}UO_2$ MOX in different pins of the same fuel cluster, and use of a heterogeneous moderator consisting of amorphous carbon and heavy water in 80%-20% volume ratio. As compared to a Pressurised Heavy Water Reactor (PHWR) the total heavy water inventory in AHWR is considerably reduced, and since the moderator heavy water operates under low pressure and temperature, loss of heavy water through leaks is practically zero, reducing spread of tritium based radioactivity. No special systems are needed for the minimisation of heavy water losses and the recovery of such losses. The negative void coefficient of reactivity considerably simplifies the burden on the reactor regulating system. Use of boiling light water coolant enables doing away with steam generator, and its substitution with steam drums of simple construction.

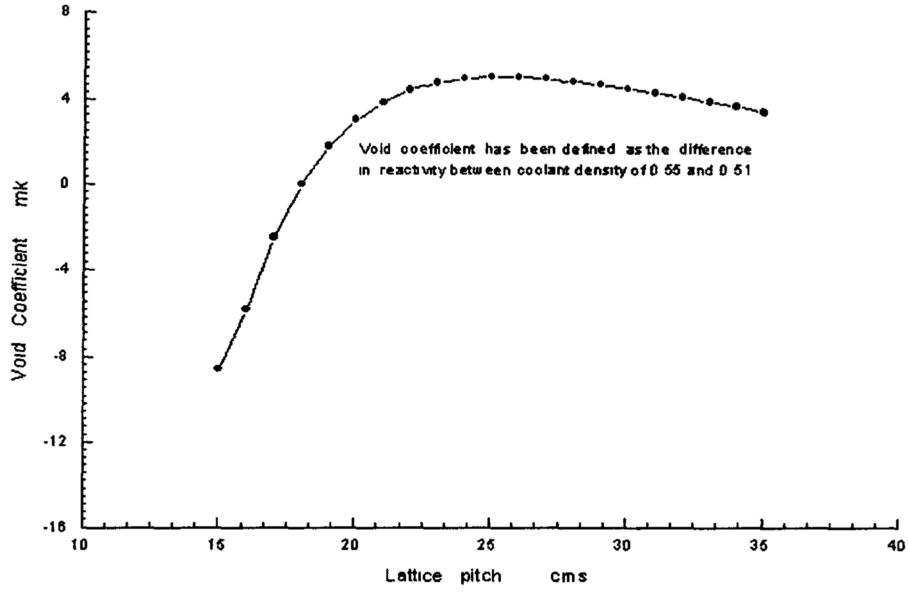


FIG. 2. Void Coefficient in the AHWR as a function of lattice pitch with heavy water moderator alone

2.2 Negative Void Coefficient of Reactivity

The ^{233}U enriched ThO_2 based fuel has a positive void coefficient, and is sub-critical, whereas the MOX fuel has a negative void coefficient of reactivity. With proper combination of MOX and ^{233}U-ThO_2 pins in a cluster it is possible to achieve overall negative void coefficient of reactivity under all operating conditions. With this inherent feature, the reactor will be shut down automatically if there is any increase in void due to any transient or accident condition. Achieving a slightly negative void coefficient for the AHWR core configuration requires a tighter lattice pitch than that required for a PHWR. This is illustrated in Figure 2.

3. COOLANT CHANNEL

3.1 Main Features

Coolant channels of AHWR serve to accommodate the fuel, maintain thermal insulation between the coolant and the cold moderator, provide interfaces for coupling to the heat transport system at the two ends of the coolant channels and provide suitable interface to facilitate fuelling. The coolant channel has suitable features to facilitate its orientation within the given lattice position, to accommodate thermal expansion and creep/growth related dimensional changes of coolant channel, to facilitate easy replacement of coolant channel, and to inject water from Emergency Core Cooling System (ECCS) in the event of a Loss Of Coolant Accident (LOCA) directly into coolant channel

A general arrangement of the coolant channel assembly, along with fuel, and some portions of end-shields, is shown in Figure 3.

The coolant channel consists of a Zirconium alloy pressure tube, a stainless steel top end fitting and a stainless steel bottom end fitting. The pressure tube is located in the core portion. The core portion is extended in both directions with the help of top and bottom end fittings. Top end fitting has suitable feature for engaging fuelling machine to the channel, and for connecting it to a tail pipe and an ECCS injection pipe. Outlet tail pipe is directly welded to top end-fitting. The feeder pipe is connected to bottom end fitting using a special flanged connection with metal C-ring as sealing element. This facilitates disconnection of bottom end fitting from the feeder pipe whenever required. The coolant channel assembly is supported on the top end shield.

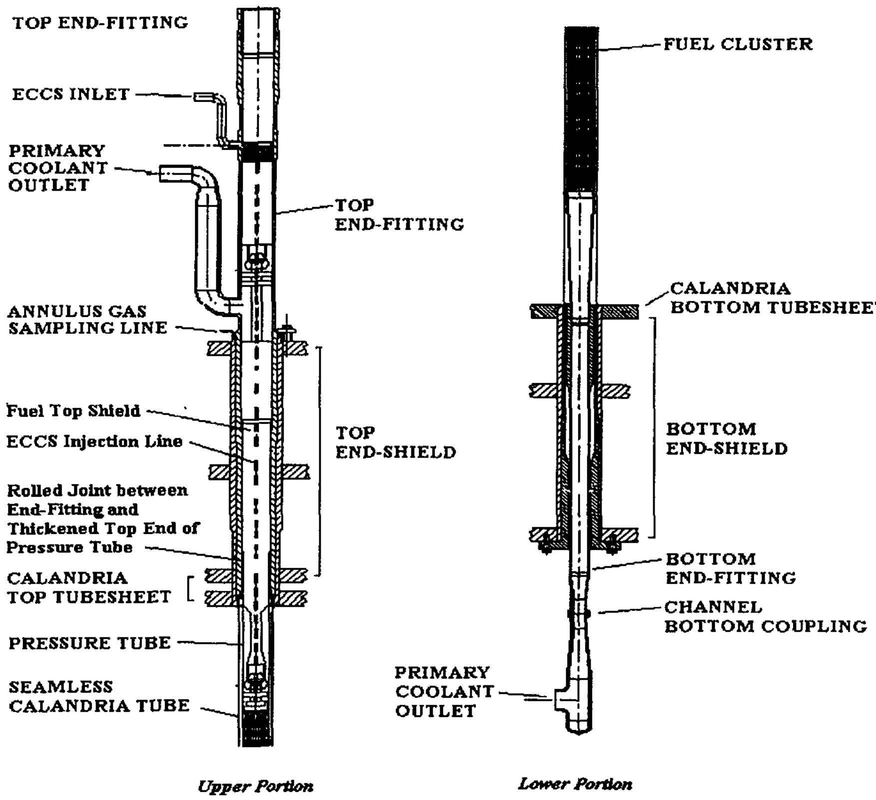


FIG. 3. Coolant channel assembly of AHWR, also showing fuel, and portions of end-shields

The fuel assembly consists of a fuel cluster and a top shield connected to each other with a collet type of joint. The coolant enters the coolant channel at 270 °C through bottom end fitting and flows through pressure tube past the fuel assembly. Coolant at 285 °C flows out through the outlet tail pipe connected to top end fitting. Typically, at the outlet from the channel the coolant has a steam quality of 15 percent. The flow of primary coolant has to occur by natural convection, hence special attention has been paid to minimise the pressure drop in the channel

3.2 Annulus Gas Monitoring

The pressure tube carrying hot coolant is insulated from cold moderator (55 - 70 °C) with an annular space formed between pressure tube and a Zircaloy calandria tube. The two ends of the calandria tube are rolled into lattice tubes of top and bottom end shields respectively. The annulus around a pressure tube and its end fittings is open at the bottom and is sealed at the top. An annulus gas monitoring system is provided to monitor for any possible leakage of heavy water either from pressure tube or calandria tube due to any failure of these tubes and their rolled joints. The air sample from annulus is sucked through a tube communicating with the annulus by using a vacuum suction device and the sample is analysed for early detection of any through crack of pressure tube or calandria tube

3.3 Easy Replaceability

On account of lower values of neutron flux and maximum operating temperature in AHWR, as compared to those in PHWRs, the rate of in-service degradation of the pressure tubes in the former will be lower than that in the latter. Even then, most of the pressure tubes of the reactor may need to be replaced at least once during the ninety year design life of the reactor. The coolant channels are designed to facilitate quick replacement of pressure tubes, during normal maintenance shutdowns.

The removal of pressure tube along with bottom end-fitting, using rolled joint detachment technology, can be done in AHWR coolant channels without disturbing the top end-fitting, tail pipe and feeder connections, and all other appendages of the coolant channel. This considerably simplifies the task of pressure tube replacement and enables a substantial saving in the cost of tooling, replacement components and downtime.

3.4 Design simplification

The channel design has been simplified by eliminating channel annulus bellows, and by having tail pipes welded to the top end-fitting. Liner tubes, existing in the end-fittings of PHWRs to guide fuel bundles, have been eliminated. The closed annulus gas monitoring system of PHWRs has been substituted by a system connected to a sniffer tube joining the top of each channel annulus, with the bottom of the annulus open to reactor cavity environment.

4. PASSIVE SAFETY FEATURES

4.1 General Description of Passive Features in AHWR

Apart from establishing a slightly negative void coefficient of reactivity, the AHWR incorporates several other passive features. These include the following:

a) Heat removal through thermo-syphon driven natural circulation under both normal operation and hot shutdown conditions.

b) Direct injection of ECCS water in fuel.

c) Passive systems for containment cooling and isolation

d) Availability of a large inventory of borated water in overhead Gravity Driven Water Pool (GDWP) to facilitate sustenance of core decay heat removal, ECCS injection, and containment cooling for three days without invoking any active systems or operator action.

4.2 Natural Circulation of Primary Coolant

During normal reactor operation, full reactor power is removed by natural circulation. The necessary flow rate is achieved by locating the steam drums at a suitable height above the centre of the core, taking the advantage of reactor building height. Figure 4 shows variation in primary flow rate with power for the design configuration of the reactor.

By eliminating nuclear grade primary circulating pumps, their prime movers, associated valves, instrumentation, power supply and control system, the plant is made simpler, less expensive, and easier to maintain as compared to options involving forced circulation in the primary coolant circuit. The above factors also lead to considerable enhancement of system safety and reliability since pump related transients have been removed. A major experimental programme has been launched to confirm

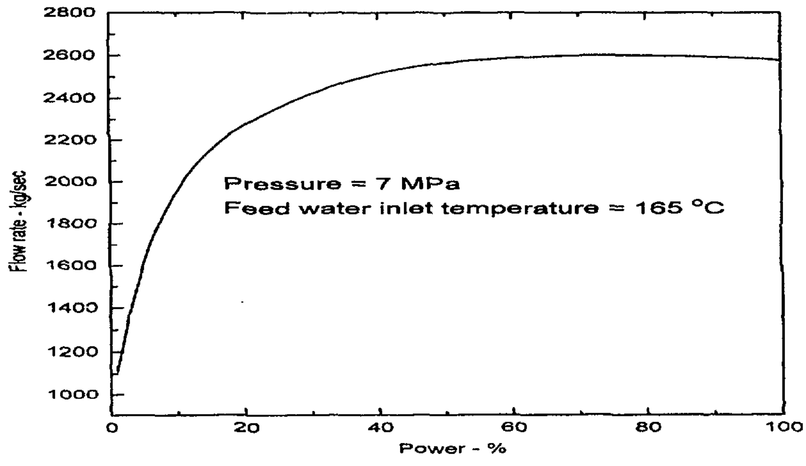


FIG. 4. Effect of power on primary flow rate

the analysis leading to the determination of loop height, and to study the thermal hydraulic stability of the Primary Heat Transport (PHT) loop.

4.3 Core Decay Heat Removal System

During normal reactor shut down condition core decay heat is removed by passive means by utilising Isolation Condensers (ICs) submerged in Gravity Driven Water Pool (GDWP) located above the steam drum. Core decay heat, in the form of enthalpy of steam, enters the IC pipe bundles through natural circulation. The steam condenses inside the pipes and heats up the surrounding pool water. The condensate returns by gravity to the core. The water inventory in the GDWP is adequate to cool the core for more than 3 days without any operator intervention and without boiling of GDWP water. A separate GDWP cooling system is provided to cool the GDWP inventory in case the temperature of GDWP inventory rises above a set value. An Active Shut Down Cooling System (ASDCS) is also provided to remove the core decay heat in case the ICs are not available.

4.4 Emergency Core Cooling System

During Loss Of Coolant Accident (LOCA) emergency coolant injection is provided by passive means to keep the core flooded so as to prevent overheating of the fuel. The emergency Core Cooling System (ECCS) is designed to fulfil the following two objectives.

a) To provide large amount of cold borated water directly into the core in the early stage of Loss of Coolant Accident (LOCA) and then a relatively small amount of cold borated water for a longer time to quench the core. This objective is achieved through use of a passive fluidic flow control device.

b) To provide water through Gravity Driven Water Pool (GDWP) to cool the core for more than 3 days.

Long term core cooling is achieved by active means by pumping water from reactor cavity to the core through heat exchangers.

The ECCS accumulators and GDWP are connected to the PHT system by rupture discs, check valves and the isolation valves kept in series. During reactor start-up, accumulators and GDWP are isolated by closing the isolation valves. When the PHT system pressure reaches the operating pressure level, these isolation valves are opened. The nitrogen pressure in accumulators is always maintained at 5 MPa to keep the system in a state of readiness. Following a postulated LOCA, when the PHT system pressure falls below 5 MPa, the rupture discs open out allowing cold borated water from accumulators to flow into the core. When accumulators get exhausted, low water level signal from accumulators results in closure of isolation valves and water from accumulator stops flowing into the core. At this stage, water from GDWP starts flowing into the core by gravity. Through an optimum

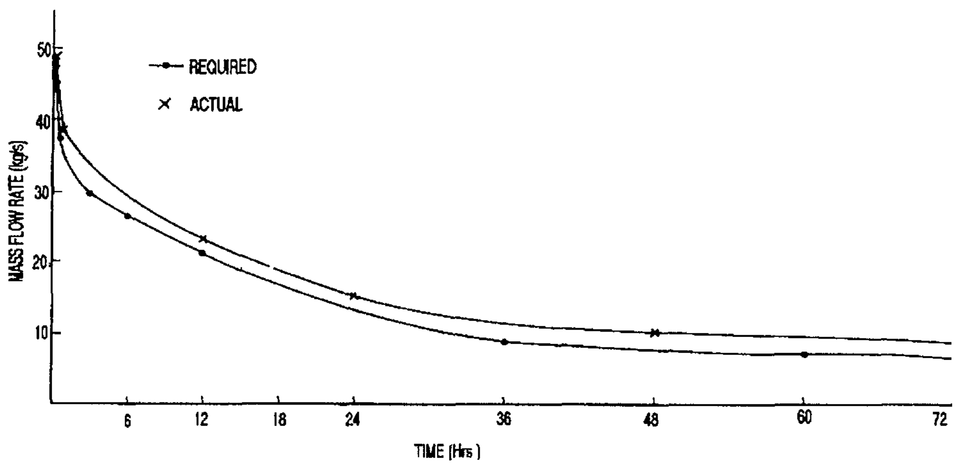


FIG. 5. GDWP flow rate for emergency core cooling

positioning of the discharge nozzles, the GDWP based ECCS flow rate is closely matched to the requirement for core decay heat removal, enabling an extended duration of availability of ECCS flow, for more than three days, as shown by the results provided in Figure 5.

After three days, water from the reactor cavity (which is filled up with hot water after spillage from ruptured pipe and water from accumulators and GDWP after cooling the core) is pumped back into the core through heat exchangers for long term recirculation mode. This heat is transferred, in the heat exchangers, to the process water which in turn dissipates its heat to the ultimate heat sink, i.e., to either sea water or to cooling tower.

4.5 Core Submergence

Following a postulated LOCA, water from the PHT system, ECCS accumulators and the GDWP, after cooling the core, will be guided and get collected in the space around the core called reactor cavity. Thus the core will be submerged under water. In the unlikely event of failure of GDWP to hold the water inventory, under any postulated scenario, the whole GDWP inventory will get collected in the reactor cavity and provide a heat sink for heat removal from the core.

4.6 Failure of ECCS during LOCA

AHWR contains, in and around its core, a large inventory of heavy water moderator and surrounding vault water. Although the possibility of failure of ECCS is very rare but if ECCS is not available during LOCA, under any postulated scenario, the fuel temperature will start rising and ballooning of pressure tubes will occur. Due to ballooning the pressure tubes will come in contact with calandria tubes and heat will be transferred to the moderator and then from moderator to vault water, thereby providing a large heat sink for the removal of core heat.

4.7 Passive Containment Isolation

For containment isolation, in addition to the normal inlet and outlet ventilation dampers, a passive system has been provided in the AHWR. The reactor building air supply and exhaust ducts are shaped in the form of U bends of sufficient height. In the event of LOCA, the containment gets

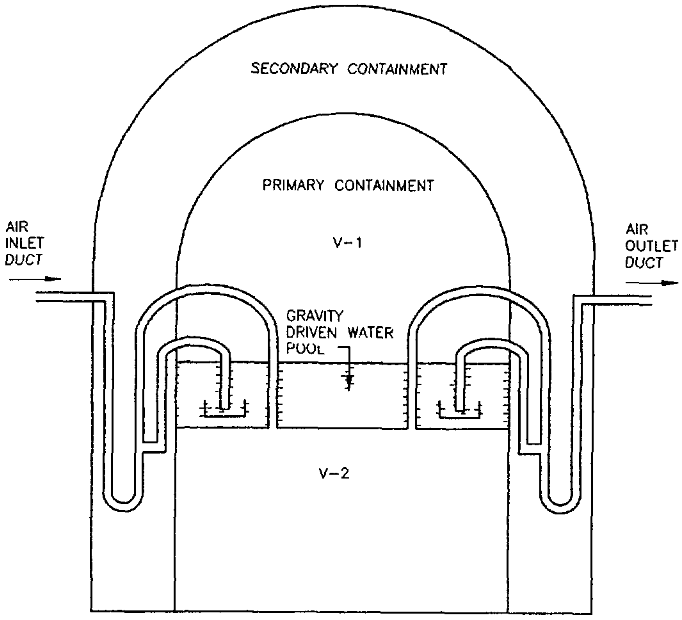


FIG 6 Schematic of Passive Containment Isolation System

pressurised. This pressure acts on GDWP inventory and pours water by swift establishment of a siphon, into the ventilation duct U bends. Water in the U bends acts as seal between the containment and the external environment, providing the necessary isolation between the two. Drain connections provided to the U bends permit the re-establishment of containment ventilation manually when desired. A schematic of this system is shown in Figure 6.

4.8 Passive Containment Cooling

Passive Containment Coolers (PCCs) are utilised to achieve post-accident primary containment cooling by passive means and to limit the post-accident primary containment pressure. A set of PCCs are located below the GDWP and are connected to the GDWP inventory. During LOCA, the mixture of hot air and steam is directed to flow over the PCCs. Steam condenses and hot air cools down at the PCC tube surface and hence provides long term containment cooling after the accident.

5. CONCLUSION

A rigorous evaluation of the feasibility of the AHWR design concept has been concluded, and its detailed design, along with a programme for conducting supporting experimental and analytical studies is underway. A detailed quantification of the economics of AHWR operation will be done when some of this work is completed. However, the economy enhancing aspects of its key design features relating to reactor physics, coolant channels and passive safety are expected to compensate for relative complexity of the thorium fuel cycle activities required to support the operation of this reactor.

WESTINGHOUSE AP600 ADVANCED NUCLEAR PLANT DESIGN

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XA0053563

Abstract

As part of the cooperative U.S. Department of Energy (DOE) Advanced Light Water Reactor (ALWR) Program and the Electric Power Research Institute (EPRI), the Westinghouse AP600 team has developed a simplified, safe, and economic 600-megawatt plant to enter into a new era of nuclear power generation. Designed to satisfy the standards set by DOE and defined in the ALWR Utility Requirements Document (URD), the Westinghouse AP600 is an elegant combination of innovative safety systems that rely on dependable natural forces and proven technologies. The Westinghouse AP600 design simplifies plant systems and significant operation, inspections, maintenance, and quality assurance requirements by greatly reducing the amount of valves, pumps, piping, HVAC ducting, and other complex components. The AP600 safety systems are predominantly passive, depending on the reliable natural forces of gravity, circulation, convection, evaporation, and condensation, instead of AC power supplies and motor-driven components. The AP600 provides a high degree of public safety and licensing certainty. It draws upon 40 years of experience in light water reactor components and technology, so no demonstration plant is required. During the AP600 design program, a comprehensive test program was carried out to verify plant components, passive safety systems components, and containment behavior. When the test program was completed at the end of 1994, the AP600 became the most thoroughly tested advanced reactor design ever reviewed by the U.S. Nuclear Regulatory Commission (NRC). The test *results confirmed the exceptional behavior of the passive systems and have been instrumental in facilitating code* validations. Westinghouse received Final Design Approval from the NRC in September 1998.

1. INTRODUCTION

The Westinghouse AP600 reactor has been designed as part of the Advanced Light Water Reactor (ALWR) Program sponsored by the U.S. DOE and EPRI. The AP600 has been reviewed by the U.S. NRC and received Final Design Approval in 1998. It is scheduled to receive NRC certification in 1999. A detailed design program (FOAKE-First-of-a-Kind-Engineering) is proceeding in parallel with the NRC certification under the sponsorship of DOE, the Advanced Reactor Corporation (ARC), and EPRI.

The AP600 is a 600 MWe reactor which utilizes passive safety features that, once actuated, depend only on natural forces such as gravity and natural circulation to perform all required safety functions. These passive safety systems result in increased plant safety and can also significantly simplify plant systems, equipment, and operation.

2. DESIGN OBJECTIVES

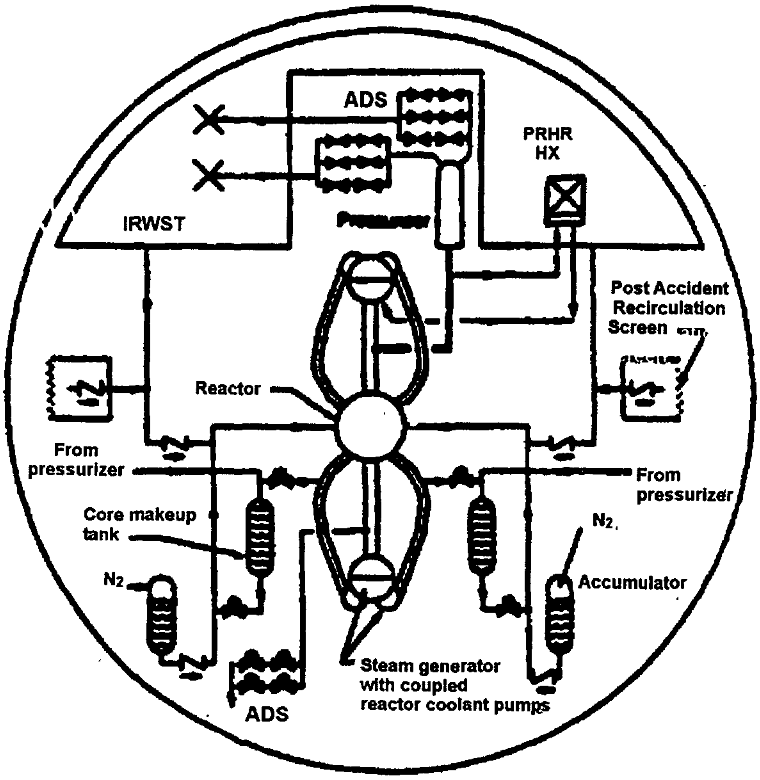
The primary design objective of the AP600 plant was to provide a greatly simplified plant design that meets the NRC regulatory requirements, meets or exceeds the NRC safety goals and ALWR Utility Requirements and addresses past safety issues while being economically competitive with other power generation systems over the full operating cycle. The objective is to be met using experience-based components so that plant prototype or demonstration models are not required. Simplification of plant systems, combined with increased plant operating margins, reduces the actions required by the operator in the event of an accident. As a result, the AP600 design requires no operator actions following a design basis accident to maintain a safe configuration. The design target for the AP600 is to technically support elimination of the emergency planning zone beyond the site boundary. The simpler systems, combined with the licensing reforms of 10CFR52, increase the licensing certainty of the AP600.

Implementation of the passive safety features greatly reduces the operation, maintenance and testing requirements of the AP600. The AP600 has been designed to have a shorter construction schedule through the use of modular construction techniques that are similar to those applied in ship construction. The construction design objective is a 36-month schedule from first concrete pour to the fuel load. An added benefit of this approach is that a significant portion of the quality assurance inspections can be completed in the factory before the modules are delivered to the construction site.

3. PASSIVE SAFETY FEATURES

The AP600 uses passive safety systems to enhance the safety of the plant and to satisfy NRC safety criteria. The systems use only natural forces, such as gravity, natural circulation, and compressed gas to make the system work. No pumps, fans, diesels, chillers, or other rotating machinery are used. A few simple valves are used to align the passive safety systems when they are automatically actuated. In most cases these valves are "fail safe" (i.e., they require power to stay in their normal, closed position; loss of that power causes them to open to their safety alignment. This power is normally supplied by class 1E uninterruptible power supplies). The passive safety systems are significantly simpler than typical PWR safety systems.

In addition to being simpler, the passive safety systems do not require the large network of safety support systems needed in typical nuclear plants, such as AC power, HVAC, and cooling water systems and seismic buildings to house these components. This simplification includes eliminating the safety-grade emergency diesel generators and their network of support systems, air start, fuel storage tanks and transfer pumps, and the air intake/exhaust system. As a result, the support systems no longer need to be safety grade and can be simplified or eliminated.



IRWST - In-Containment Refueling Water Storage Tank
PRHR HX - Passive Residual Heat Removal Heat Exchanger
ADS - Automatic Depressurization System (4 stages)

FIG. 1. Schematic Representation of the In-Containment Passive Safety Injection System

The features of the AP600 passive safety systems include passive safety injection, passive residual heat removal, and passive containment cooling. All these passive systems have been designed to meet the NRC single-failure criteria and its recent criteria including TMI lessons learned and unresolved and generic safety issues. PRAs have also been used to quantify the safety of the design.

3.1. Passive safety injection system

The passive safety injection system (PSIS), Figure 1, performs three major functions: residual heat removal, reactor coolant makeup for inventory control, and safety injection. Computer analyses demonstrate that the PSIS provides effective core cooling for various break sizes and locations. These calculations show that the PSIS prevents core damage for breaks as large as the 0.2m (8 inch) vessel injection lines and provides about 260°C (500°F) margin to the maximum peak clad temperature limit for the double-ended rupture of a main reactor coolant pipe.

The passive residual heat removal heat exchanger (PRHR HX) protects the plant against transients that upset the normal steam generator feedwater and steam systems. The analysis results, using NRC-approved codes, has shown the PRHR HX to satisfy the NRC safety criteria for loss of feedwater, feedwater line breaks, and steam line breaks with a single failure. Anticipated transients without reactor trip have also been analyzed and shown to result in peak RCS pressures of about 20 MPa (2900 psig), well within NRC criteria. The PRHR HX consists of a 100 percent capacity bank of tubes connected to the RCS in a natural circulation loop. The loop is normally isolated from the RCS by valves that are normally closed, but fail open if power is lost. The heat exchanger tubes are located in the in-containment refueling water storage tank (IRWST). This location places the PRHR HX above the RCS loop so that hot water leaving the RCS hot leg will rise to the top of the PRHR HX where it is cooled. The difference in temperature between the hot inlet water and the cold outlet water drives the natural circulation loop. If the reactor coolant pumps are running, they boost the PRHR HX flow.

The IRWST provides the heat sink for the PRHR HX. The IRWST water volume is sufficient to absorb decay heat for about 2 hours before the water would start to boil. After that, steam would be generated and enter the containment. This steam would condense on the steel containment vessel and then drain back into the IRWST.

The PSIS uses three sources of water to maintain core cooling, including core makeup tanks (CMTs), accumulators and the IRWST. All of these injection sources are connected directly to two nozzles on the reactor vessel. These connections, which have been used on existing two-loop plants, reduce the possibility of spilling part of the injection flow.

Passive reactor coolant makeup is provided to accommodate small leaks following transients or whenever the normal makeup system is unavailable. Two CMTs, filled with borated water, are designed to provide this function at any RCS pressure using only gravity as a motivating force. These tanks are designed for full RCS pressure and are located above the RCS loop piping. If the water level in the pressurizer reaches a low-low level, the reactor is tripped, the reactor coolant pumps are tripped, and the CMT discharge isolation valves open automatically.

The relative elevations of the CMTs and the pressurizer are such that if RCS level continued to decrease, the water in the CMTs would drain into the reactor vessel.

As with current pressurized water reactors (PWRs), high pressure accumulators are required for large loss-of-coolant accidents (LOCAs) to meet the need for higher initial makeup flows to refill the reactor vessel lower plenum and downcomer following RCS blowdown. The gas pressure forces open check valves that normally isolate the accumulators from the RCS. The accumulators are sized to respond to the complete severance of the largest RCS pipe by rapidly refilling the vessel downcomer and lower plenum. The accumulators continue delivery to assist the CMTs in rapidly reflooding the core.

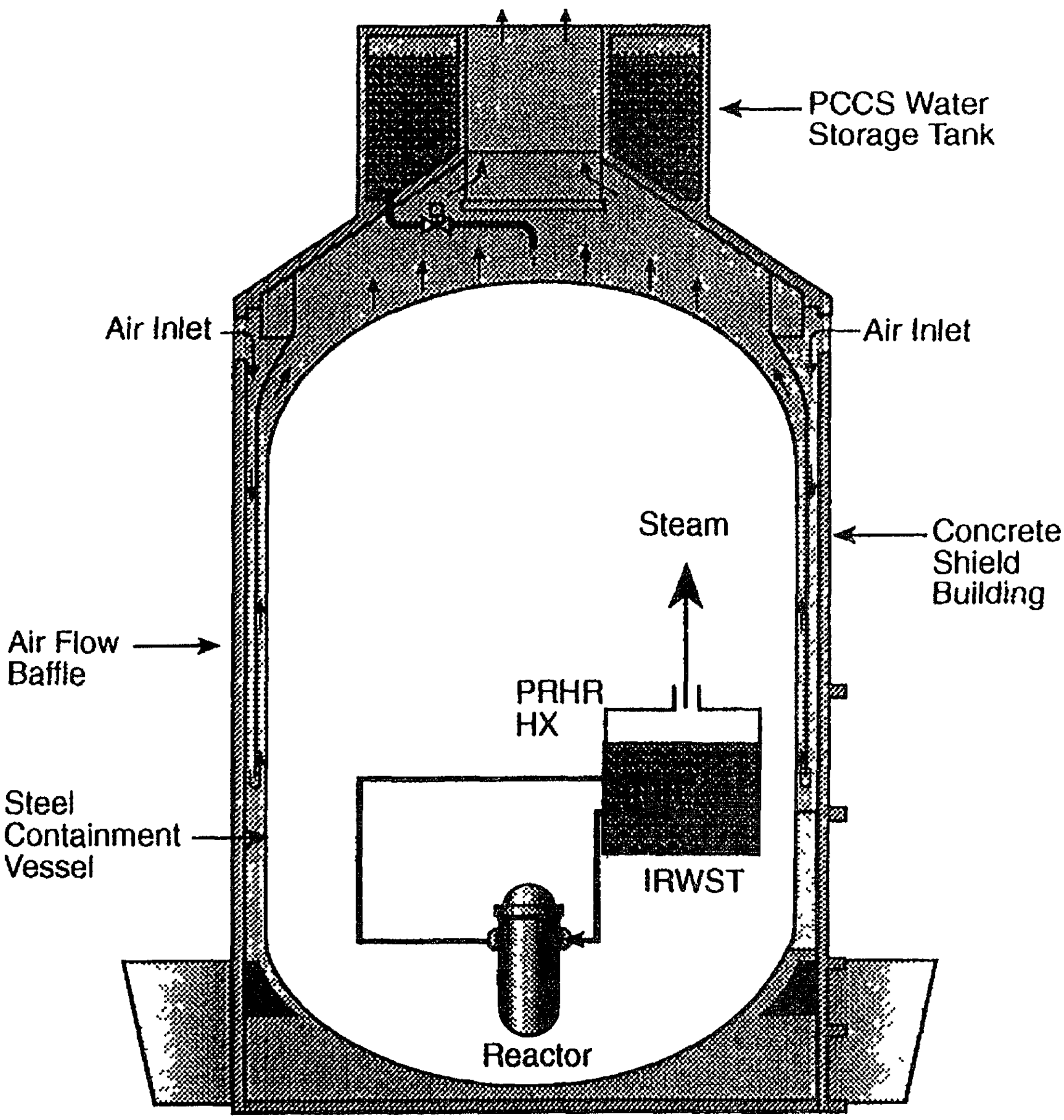


FIG. 2. AP600 Plant Heat Sink

Long-term injection water is provided by gravity from the IRWST, which is located in the containment just above the RCS loops. Normally, the IRWST is isolated from the RCS by self-actuating check valves This tank is designed for atmospheric pressure. As a result, the RCS must be depressurized before injection can occur. The AP600 automatically controls depressurization of the RCS to reduce its pressure to about 6.89 x 10^4 Pa (10 psig), at which point the head of water in the IRWST is sufficient to overcome the small RCS pressure and the pressure loss in the injection lines. The automatic depressurization system (ADS) is made up of four stages of valves to permit a relatively slow, controlled RCS pressure reduction The first three stages are connected to the pressurizer and discharge through spargers into the IRWST. The fourth stage is connected to a hot leg and discharges through redundant isolation valves to the containment. The ADS stages are actuated by CMT level.

3.2. Passive containment cooling system

The passive containment cooling system (PCCS) provides the safety-related ultimate heat sink for the plant, as shown in Figure 2. As demonstrated in the computer analysis and tests, the PCCS is able to effectively cool the containment following an accident such that the design pressure is not

TABLE I. AP600 PRA RESULTS

Event	AP600	Current Plants
Transients	5.0E-9	1.3E-5
Loop/Blackout	1.0E-9	6.6E-6
SG Tube Rupture	6.1E-9	1.7E-6
LOCA - Small	1.0E-8	8.0E-6
LOCA - Medium	7.6E-8	5.0E-6
LOCA - Large	5.0E-8	8.0E-7
ATWT	1.0E-8	2.2E-6
Loss of Cooling	<E-9	1.1E-5
Interfacing LOCA	<E-9	1.0E-6
Vessel Rupture	1.0E-8	3.0E-7
Total - Safety and Nonsafety Systems	1.7E-7/Yr	5.0E-5/Yr
Safety Systems Only	7.7E-6/Yr	

exceeded and the pressure is rapidly reduced. The steel containment vessel itself provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Heat is removed from the containment vessel by a natural circulation flow of air that cannot be isolated. During an accident, the air cooling is supplemented by water evaporation on the outside of the containment shell. The water is drained by gravity from a tank located on top of the containment shield building. Two normally closed, fail-open valves are opened to initiate the water drain. The water tank is sized for three days of operation, after which time the tank is expected to be refilled to maintain the low containment pressure achieved after the accident. If the water is not resupplied after three days, the containment pressure will increase, but the peak is calculated to reach only 90 percent of design pressure after about two weeks.

3.3. Severe accident considerations

PRA has been an integral part of the AP600 design process. The AP600 PRA report was a joint effort between Westinghouse and Ente Nazionale per l'Energia Elettrica of Italy. Numerous design changes were made as a result of examination of the PRA results, for example, the diversity in the fourth stage ADS valves, logic in the startup feedwater system and active valves in parallel with several of the key check valves. The resulting core damage frequency for the AP600 is 1.7 E-7/yr. This compares well to the NRC safety goal of 1.0 E-4/yr and the ALWR goal of 1.0 E-5/yr. Typical numbers for current plant designs are 5.0 E-5/yr.

The PRA also illustrates that the AP600 containment design is robust in its ability to prevent releases following a severe accident and that the risk to the public due to severe accidents for AP600 is very low. The overall release frequency for AP600 is 1.8E-8/yr. This meets the NRC safety goal and the ALWR goal of 1.0E-6/yr. Typical numbers of current plant designs are 5E-6/yr.

The PRA analysis shows that the capability to flood the reactor cavity prevents the failure of the reactor vessel given a severe accident without water in the cavity. The vessel and its insulation are designed so that the water in the cavity is able to cool the vessel and prevent it from failing (termed in-vessel retention of molted core debris). Maintaining the vessel integrity eliminates the potential of a large release due to ex-vessel phenomena and its potential to fail the containment.

By reducing the reactor system pressure, the ADS eliminates the possibility of high pressure core melt ejection and the resultant direct containment heating. The AP600 also has an igniter system to mitigate the effects of hydrogen released during a severe accident.

4. TEST PROGRAM

The AP600 is, of course, designed to the last function and flow rate with computers. According to the computer codes, each of the AP600's systems works even better than required. The primary burden of the extensive testing program has been to supply hard data to verify those computer codes.

To perform the testing, some of for the best facilities in the world were selected. Work was performed at Oregon State University; at a thermal-hydraulic test complex and a steam testing facility in Italy; in Ontario, Canada, at the University of Western Ontario; and at two of Westinghouse's technology development centers in Pittsburgh.

The Oregon State work was the "crown jewel" of the testing program. The work was conducted in a new facility. The tests went so well that the work planned for several months was performed in just six weeks, completing the last test on August 3, 1994. The Oregon State testing focused on 30 small-break loss-of-coolant accidents (LOCAs). Using a one-quarter scale model that replicates the AP600, the transition from LOCA events into long-term cooling were simulated, relying on coolant injected from the passive safety systems. The tests validated the codes, proving that the AP600 core will be adequately cooled at all times during all LOCAs.

Other LOCA tests were conducted in Piacenza, Italy, at the SPES-2 facility. These were full-height, full-pressure simulations of both the primary cooling and passive core cooling systems. Thirteen small-break LOCAs were conducted, as well as simulations of ruptures in steam generator tubes and steam lines. Again, the results validated the safety analysis computer codes and models.

In Casaccia, Italy, full-scale, full-flow tests were done on the automatic depressurization system. These tests went through two major phases -- first, 21 steam blowdowns, and second, 24 steam and water blowdowns. And again, the results matched the computer modeling.

In London, Ontario, Canada's University of Western Ontario used its wind tunnel to be sure that winds -- especially high winds -- would not diminish cooling in the annulus of the containment while the natural convection and evaporative forces were doing their job. A detailed scale model of the AP600 was placed in the tunnel. The Ontario tests shows that wind was not a safety factor. The natural circulation of air occurs over the surface of the reactor shield under all wind conditions.

At the Westinghouse Science and Technology Center in Pittsburgh, containment cooling work was performed. First, a flat plate was built ... 0.91 m (3 feet) wide and 1.82 m (6 feet) high ... that met the specifications for the containment. It was used it to examine basic thermodynamics. Water was run across its surface to acquire hard data on the heat transfer to be expected under varying circumstances. The data took us a step forward in the containment code safety analysis.

A much larger test device was built -- a 7.3 m (24-foot) high, 0.91 m (three-foot)-diameter model of the containment. This time, steam was put inside, and air moved over the outside as water ran across the surface. This larger scale, more detailed testing gave us even more data for the computer codes. Then an even larger model was used: a one-eighth scale model of the containment. Again, the entire range of passive safety actions were tested -- using various internal conditions and external air and humidity values. Tests were conducted with water applied to the top of the external surface, and with a dry surface. This exhaustive, real-world testing shows that the theory of passive containment cooling is valid. Even the toughest internal and external conditions -- and combinations of both -- did an adequate job of cooling. While water applied to the top of the vessel helped, it was determined that it was not absolutely needed, even from the first moment of an event. This is a cooling concept that just sits there and does the job perfectly, relying on nature.

At another Westinghouse test facility, a scaled version of the core makeup tank was constructed. It was used to investigate the thermal-hydraulic behavior of the tank under a wide range of conditions. The data from this series of tests was a major source for the refinement of the computer model codes.

Those are just the highlights of the testing. Along the way at these sites, the reactor coolant pumps, check valves and incore instrumentation were tested as well as a wide variety of flow and heat transfer tests. Every aspect of the passive safety system was simulated -- from safety-grade components to the containment cooling and core cooling systems.

The testing was thorough and exhaustive, but the results were well worth the cost and effort. The AP600 is now the most thoroughly tested reactor design ever reviewed by the NRC. The very sophisticated, detailed, conservative computer codes were validated which are used to analyze the AP600. In some cases, the data was used to refine the codes. The data and codes predict safety under a long list of scenarios, both normal and abnormal. The AP600 is safe under all predicted accident conditions. These tests met the very demanding requirements defined at the start by the DOE, the NRC, and the supporting utilities.

5. LICENSING STATUS

One of the most challenging aspects of the testing was the initial agreement on the scope and nature of the tests. All of this work has been monitored very closely by both the DOE and the NRC through a series of milestones. This steady advance in testing has now brought the AP600 to the equally painstaking process of licensing.

In June of 1992, the safety analysis report and probabilistic risk assessment report was submitted to the NRC. The Commission replied with a draft safety evaluation report, listing its questions. Westinghouse responded to them. In some cases, the response created open issues that we have resolved. In May of 1996, the NRC issued a supplement to the Draft Safety Evaluation Report on the AP600 computer codes and testing program. This report identified the final questions the staff had concerning the computer codes and testing program. Westinghouse has responded to the last of these questions.

This highly technical iteration continued through 1997. Westinghouse has now resolved all the issues to the NRC's satisfaction. In May 1998, the NRC's technical staff approved the Final Safety Evaluation Report for the AP600. Final Design Approval (FDA) was received in September 1998, following review by both the Advisory Committee on Reactor Safety (ACRS) and the NRC Commissioners. The FDA documented NRC acceptance of AP600 safety and at this point, the AP600 is a saleable plant on international markets. In the United States, there will be an additional year for a public rule-making, after which the NRC will be able to issue a final design certification. This final design certification is relevant only in the United States and is not needed in other countries.

Westinghouse now has what no other advanced passive plant has: the numbers on performance, on safety, on costs, and on schedule. Westinghouse has told the utility supporters that it can meet their goals.

Financially, the AP600 will do better than the original cost goals established by America's utilities for a 600-megawatt advanced passive plant. A twin-unit site can meet the tougher cents-per-kilowatt-hour goals established for a 1200-megawatt plant. The AP600 can compete with the largest plants now being designed -- whether nuclear or coal.

The AP600 defies assumptions about economies of scale because its initial costs are well down that scale. That is the value of safety through simplicity. It is the value of the 600-megawatt size and three-year timetable for construction. The challenge of financing an AP600 is relatively simple. This is a plant that can be built ... starting now.

KEYS TO ECONOMIC VIABILITY OF EVOLUTIONARY WATER COOLED REACTORS

(Session V a)

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LESSONS LEARNED FROM STANDARDIZED PLANT DESIGN AND CONSTRUCTION

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Abstract

Following France's hydropower program, Electricité de France began producing electricity with nuclear power during the 60s with units that were all different. In the early 70s, EDF launched an extensive nuclear program which was fueled by the 1973 oil crisis. The particularity of this program is based on the standardization of the design which enables the cost of engineering studies, components and construction to be reduced. As all of the sites presented various conditions, a single design was possible except for the heat sink, connection to the grid and foundations. In order to follow technical progress, the program was divided into several homogeneous series: CP0, CP1 and CP2 for 900 MWe reactors, P4 and P'4 for 1300 MWe reactors and N4 for 1450 MWe reactors. EDF has managed to apply standardization throughout the service life of the plant: all units of the same series are modified in the same manner and with a same batch of modifications. The standardization of operations is also the EDF's rule: technical specifications, safety reports, and safety procedures are normally the same for units belonging to the same series. Nevertheless, when plant design and operations, and heavy maintenance are considered, it becomes increasingly difficult to maintain strict standardization across the board: when examined closely, variations are possible—as regards the chemical specifications of the secondary system, for instance. On the other hand, at the fabrication stage, it is difficult to maintain fabrication procedures and alloy compositions rigorously the same. Standardization offers a tremendous advantage representing 30-40% of construction costs. The main drawback is the risk of generic defects. On the other hand, the risk is rather small owing to the small differences among units.

1. CONTEXT OF THE FRENCH NUCLEAR PROGRAM LAUNCH

1.1 Historical reminder

During the fifties and the early sixties, EDF (Electricité de France) began an ambitious hydro-electric program with the construction of numerous dams at available sites: during this period, EDF (Direction de l'Equipement) developed its experience as an industrial architect; the national company actively participated in the general project design, the establishment of contracts with subcontractors (civil works, electromechanical components etc.) and the overseeing of construction operations.

In the second half of the sixties, the number of sites available for hydroelectric power production dwindled. Within the scope of its continued efforts toward diversification, EDF decided to turn to classical thermal (fuel) and to nuclear power generating plants involving a variety of different reactor types: pressurized water (Chooz A, connected to the grid in 1967, in cooperation with the Belgians), heavy water (Brennilis, connected to the grid in 1967), natural uranium gas-cooled (St Laurent 1 and 2, Chinon 2 and 3, Bugey 1 commissioned in 1969, 1971, 1965, 1966, and 1972, respectively) and fast neutron (Phénix, connected to the grid in 1973). These latter 3 reactor types were developed by the CEA "Commissariat à l'Energie Atomique" which gained experience with the Marcoule G1, G2 and G3 gas-cooled reactors (connected to the grid between 1957 and 1960, and intended for the production of electricity and fissile material) and with the Rapsodie fast neutron reactor. It should be pointed out that EDF's gas-cooled plants were essentially all different which lead to difficulties related to non-standardization (negative influence on costs, ease of operation, etc.).

1.2 The oil crisis of 1973 and its consequences

The gas-cooled plants quickly demonstrated their technological (graphite stacking fragility, low reactivity, sensitivity to the Xenon and Samarium effect) and economical limitations (lack of standardization).

The French PWR program really took off in 1969 with the Tihange 1 Franco-Belgian unit. After that, construction of 1 or 2 PWR 900 MW pre-series units began each year from 1970 on at the Fessenheim and Bugey sites.

The 1973 oil crisis, with the oil embargo and the quadrupling of oil prices, clearly reminded France of its dependence and vulnerability due to its relatively low reserves of economically exploitable oil, gas and coal. In 1973, its dependence on foreign energy sources had reached 75%.

Nuclear power was the only possible means of reducing this dependency. In early 1974, the French government decided to undertake an extensive power generation program: 13,000 MW in 1974 and 1975, 12,000 MW in 1976 and 1977 and 5,000 MW each year from 1978 to 1981. A crucial decision was made in 1975 to concentrate this program on PWRs and to launch a four-loop, 1,300 MW series.

2. ORGANIZATION FOR THE DEVELOPMENT OF THE FRENCH NUCLEAR PROGRAM

EDF decided to become deeply involved in the process by thoroughly implementing its ability as an industrial architect which it developed during the hydro and classical thermal power generation programs (with its various production levels: 125, 250, 600 and 700 MWe for the thermal plants).

Power plants are not ordered "on a turn-key basis". The plant is divided into sections including design, electromechanical and civil works engineering sub-assemblies. In this manner, EDF can open each of these sections up to the competition (except for the main primary system and its connecting circuits handled exclusively by FRAMATOME and the turbine generator plant reserved for ALSTHOM). This enables it to benefit from competitive pricing and to set up a long term partnership with the contractor retained for each section. In an extremely limited number of cases, the contract was split between two contractors; this has occurred particularly for the reactor building containment, for the installation of piping in the reactor building of 900 MWe units, depending on whether the unit had an even or odd number or for the cast elbows of the primary circuit. In exchange, this type of allotment system requires excellent control of the entire plant's operation—in short, the undeniable qualities of an architectural engineer. It should be pointed out that French nuclear safety regulations require the operator to defend its safety reports itself which has led EDF to invest in the plant process.

From these premises, we can conclude that EDF had to acquire the conceptual design of the plant, come up with the detailed design of the systems with the help of subcontractors, monitor the construction operations and defend the reports before the Safety Authorities.

The economical interest of the allotment concept was reinforced by the standardization policy which was the cornerstone of the French nuclear power program.

3. THE CONCEPT OF STANDARDIZATION FROM DESIGN AND INSTALLATION TO OPERATION

EDF's choice consisted in trying to maintain the concept of standardization throughout the plant's service life.

3.1 Standardization of engineering and design

A review of the various possible sites showed that it was possible to work out standard design basis rules which would be accepted by the French Safety Authorities at all locations.

A nuclear program developed at several sites over a period of a few decades comes up against two main types of limitations:

a) each site is characterized by specific conditions relating to geology, seismicity, human geography, industrial environment (for example, the risk of explosions of inflammable products stored close by or the dangerous proximity of public thoroughfares), by a determined heat sink (flow, chemical quality, etc.) and by a neighbouring power grid with its specific characteristics.

   It would have been anti-economical to group all site conditions together to define a design standard; the choice made consists in defining a standard which could be adapted to each of the sites with a minimum of modifications. In practice, this was possible and only the foundations, the cooling water systems and the connections to the power transmission network differ from one site to another: nearly all conventional or nuclear island systems are strictly identical as concerns the various units of the same series. The components are identical, their general manufacturing specifications and their installations are the same except for possible differences in symmetry for units built in pairs (Fessenheim, Bugey or those included in program contract No. 1). In this case, we transform the drawings of an even plant unit into the drawings for an odd plant unit by simply turning the drawing.

b) The second absolute limitation of a design standard concerns technological developments. Should the design be stabilized over a long period of time in order to be able to fully benefit from standardization or otherwise allow for technical improvements or appropriate modifications of safety and design basis rules? The compromise that was agreed upon consisted of having different plant series: 4 standardized plant series for the 900 MWe reactors (Fessenheim, Bugey, CP1 or contract program No. 1, CP2 or contract program No. 2), 2 series for the 1300 MWe reactors (P4 and P'4) and a single plant series, up until now, for the N4 1,450 MWe reactor.

For example, in comparison with Fessenheim, the Bugey units have a higher thermal output (from 2660 MW to 2785 MWe) and an enhanced containment design pressure (4.7 to 5 bar). Shifting from the "Bugey" design to the "CP1" design, we notice an increase in the height of the reactor building and an equipment hatch which rises from ground level to mid-height of the reactor building containment. Continuing on from CP1 to CP2, the turbine building which was used for a pair of units and parallel to the axis formed by the centers of 2 paired reactor buildings is split, each turbine building being perpendicular to the axis formed by the centers of both reactor buildings (the major interest of this modification is to avoid the risk of missiles coming from the turbo-generator vis-à-vis the reactor).

Simply speaking, the equipments inside plants of the same series are identical, and they are laid out in the same manner. The only differences concern the foundations (for example, at Cruas, considering the site's seismicity, the foundation raft had to be installed on aseismic bearing pads), the heat sink (certain units are cooled directly by the river, others use sea-water or cooling towers) and the connection to the power grid.

Additional aspects must be discussed if a more in-depth analysis is to be considered: within the same series, feedback may lead a builder to modify a chemical composition or a manufacturing process. A good example concerns the elbows of the main primary system: during the construction of the P4 plant, the influence of molybdenum on the in-service behaviour of the elbows became apparent and the composition of the alloy used in this element had to be modified; elbows which are more or less sensitive to the effects of aging can thus be found within the same series. Furthermore, when two manufacturers supply the same item, even though the general manufacturing specifications provided by EDF are the same, the manufacturing processes can be slightly different which leads to varying in-service behaviour. A good example of this concerns the steam generator tubes or the cast elbows of the reactor coolant system (manufactured by FAM or Creusot Loire).

Another example of de-standardization can be seen with the aggregates used in the fabrication of concrete. For obvious reasons the aggregates are taken from land surrounding the site. Their characteristics thus varies from site to site which explains why certain containment buildings experience creeping problems while others of the same series do not.

The advantages of standardization of the design are obvious:

- engineering manpower for design, quality control and assurance totals about 5 million man-hours for a first-of-a-kind unit and less than 1 million man-hours (basically attributed to site adaptation studies) for the following identical units.
- lessons learned at all stages of field work save both time and money; for instance, field manpower was about 29 million man-hours for the first two pairs of units at the Gravelines power station, and only 13 million for the third pair. Although more difficult to assess, fabrication costs most certainly benefit from standardization also.
- the allotment system, in conjunction with standardization, incites competition among suppliers resulting in the best prices for components, and construction tasks. Subcontractors are able to plan their work schedule on a long term basis, which helps reduce costs.

In summary, it can be seen that standardization of the design explains why French nuclear power plants are less expensive than foreign plants: 30-40% of the construction cost difference can be explained by standardization.

Standardization is also advantageous in terms of the execution time table. Consider the 900 MWe series, for example: for the earlier Fessenheim and Bugey units, the delivery time from vessel commitment to first connection to the grid was between 75 and 80 months, but then dropped to between 55 and 65 months for the last 10 plants (of the "900 MWe" series) connected to the grid.

3.2 Standardization and backfitting policy

The policy of successive series which ensues from technological progress and the need to improve safety and the operability of units does not exclude that the units already built are back-fitted, on the contrary. It is under this condition that the French Safety Authorities accepted the "per series" standardization principle. These back-fitting operations must not challenge the general plant design and must not be carried out at frequent intervals: the Safety Authority has informed EDF that plant back-fitting operations shall now be carried out on a ten-year basis (this period providing a certain stability within the safety reference frame). In the past, the frequency of modifications was variable, but EDF has and will always ensure that these modifications are carried out in "batches".

For instance, after TMI and in the early 80's, some improvements were required on the man-machine interface, such as control-room ergonomics and the introduction of a safety-panel. These modifications were implemented in all 900 MWe units. A systematic review of valves was also carried out and led to the decision to change the pressurizer relief valves on all units.

Another batch was commissioned in 1990, which corresponded to the "end of series" for the 900 MWe series and provided enhanced protection against the risks of boron dilution. A third batch was launched in 1993. This batch provided for automatic water makeup on the reactor residual heat removal system in shutdown condition. A new "VD2" batch ("2ème visite décennale", 2^{nd} ten-yearly in-service inspection) will be implemented on the units for their 20^{th} year in service. With each new batch, the unit's safety analysis report is updated; it should be recalled that each unit benefits from a standard safety analysis report relative to the series and a safety analysis report which specifically concerns the site.

In practice, a batch of back-fitting operations on a series is carried out over several years: thus, at any given moment, all of the units within the same series are not all identical. This once again is a limit to strict standardization.

In addition to back-fitting operations which are derived from operational feedback and which leads to an improvement of safety-operability combination, mention should be made of the back-fitting associated with generic defects within the park in operation, and the negative consequences of standardization. The best examples concern the corrosion of steam generator tubes or the cracks on the vessel heads detected at control rod mechanism penetrations. In the case of the vessel heads, EDF decided to replace all covers of 900 and 1300 MWe reactors (at a cost of approximately 50MF per reactor); the stress corrosion of the Inconel 600 used was at the heart of the problem. In the case of the steam generator tubes (for which the replacement cost is higher, at approximately 600 MF per unit), all of the 900 and 1300 MWe reactors are potentially concerned due to the sensitivity of Inconel 600; in fact, all of the steam generators will not have to be replaced. The mounting of tubes onto the tube sheet (expansion) is not strictly the same between the various units: the limits of standardization that were highlighted above (the succession of manufacturing processes is not identical) protected us from generalized problems. Curiously enough, the Fessenheim steam generators were affected to a lesser extent than certain steam generators of more recent units (technical progress is not always positive).

In practice, EDF recorded relatively few generalized generic defects at all units. This success is explained by the initial choice of a proven design (the 900 MWe plants had an American "reference plant") and the series policy (benefiting from the feedback, although possibly negative, from earlier series).

For the last few years the yearly expenditures for back-fitting and major maintenance operations have been limited to a total of approximately 4 billion francs. Considering that this sum covers ten or so units, and if we add the expenses from the past, we see that the back-fitting expenses represent less than 10% of the building cost. This sum does not question the beneficial effects of the standardization policy which can be evaluated (see Chapter 3.1) at roughly 30% of the construction cost of the units.

3.3 Operation standardization

The standardization policy also applies to plant operation, particularly for that which concerns matters of nuclear safety: it should be recalled that for nearly all systems, the applicable safety analysis report is the series' generic report.

In this manner the general operating rules, the technical specifications and the operating instructions are, in theory, identical for all the units in a series. This ensures proper homogeneity of operations within the same series; the advantages of such a policy are obvious:

- easy integration of operational feedback between identical units,
- personnel can move from one unit to another of the same series without any special training, thereby enhancing labour mobility (and better flexibility),
- a reduction in the required number of control simulators,
- control error risks are reduced, thereby enhancing safety,
- the work necessary for the drawing up instructions and procedures is lightened,
- reduction in the number of replacement parts in storage,

and of course, all of these advantages are conducive to lower operating costs.

Be that as it may, standardization policy limitations appear at this point more than at the design or back-fitting stage. This can be explained by the following points:

- due to the importance of nuclear power (more than 75% of French power production is nuclear), French units are required to follow the grid. The number of outages, loading or load reductions, and emergency shutdowns differs from one unit to another. The mechanical and thermal loads on the equipment (and thus its service life) differ from one unit to another of the same series;

- management of the fuel and its supplier may vary among units of the same series which could bring about different product behaviors;
- finally, within common technical specifications the operator has a certain amount of manoeuvrability. As such, the chemistry of the secondary system and the maintenance policy may be slightly different from one site to another. This could explain different degrees in corrosion or wear of equipment among sites.

All these factors of dissimilarity between units of the same series have a rather favourable influence particularly in terms of possible generic defects. We have already mentioned that these differences could explain why certain steam generators of similar design and equipment have to be replaced in certain plants and not in others.

3.4 Standardization and feedback of experience

An organization has been set up so that experience gained during operation is fed back in the design, construction and commissioning of the projects. Therefore, faults revealed during testing and operation of the first units (control rod guide tube pin failures, superheater component deformations, etc.) have not only been repaired in the defective units, but have been the subject of preventive measures for the units under construction. Each operating event is analyzed and may lead to a corrective modification. Similar treatment is given to events in foreign plants, the main sources of information being the US. INPO, the OECD Incident Reporting System and the UNIPEDE USER's system. Three Mile Island follow-up actions were processed similarly.

The decision, following an operating incident or a fault, to request a modification is done by panels of experts from both the operation and the design divisions.

4. PERSPECTIVES FOR THE FUTURE

There is no way to undertake standardization if safety requirements are continually modified. This obvious proposition has led Electricité de France and Framatome to become active partners of the Safety Authorities, and even to promote improvements when they appear technically useful. As far as they were concerned, the French Safety Authorities soon accepted that standardization presented a certain number of advantages from the standpoint of safety that the way was open to compromise.

As for Electricité de France, the standardization policy allows nuclear kilowatthours to be produced at a particularly competitive price and the future PWR 2000 series intended to replace present-day plants when the time comes will most certainly be built within the scope of standardized guidelines.

The success experienced in France has been a source of inspiration for other countries: in the early nineties, the United States launched the ALWR (Advanced Light Water Reactor) program which produced a set of specifications (U.R.D.: Utilities Requirement Document) relative to the construction of nuclear plant units with passive or evolutionary reactors and approved by the NRC. The program obtained the certification of a limited number of reactors (System 80+ by Combustion Engineering, ABWR by General Electric, AP600 by Westinghouse), which should pave the way for the construction of standardized models in the United States or elsewhere in the world.

Furthermore, utilities from 10 European countries (Sweden, Finland, Germany, Belgium, the Netherlands, the United Kingdom, Spain, Italy, Switzerland and Russia) have come together with EDF to create the "European Utilities Requirements" which are specifications for the construction of nuclear reactors in Europe. This initiative should lead to a limited number of models being placed on the market. Periodic exchanges have taken place between the teams responsible for drafting the URD and the EUR which enables them to focus on the general safety objectives established for future reactors with a goal toward international standardization.

5. CONCLUSION

In building its nuclear power program, EDF has implemented an extensive standardization policy encompassing design, components, back-fitting and operation throughout the service life of the plant. As a result, the units within the same series have identical components, identical layout, identical operating instructions and an identical safety analysis report (with adaptations for the specific site). The advantages of such a program are obvious: lower engineering costs, lower component and installation costs (owing to increased competition and better manufacturing scheduling), enhanced usage of feedback from experience, and lower spare parts and operator training costs.

The inconveniences of standardization are essentially limited to the generic defects, although they are not as extensive as one may be lead to imagine: there are different series, on the one hand, and on the other hand, if we examine the situation closely, in reality two units of the same series do not necessarily have the same operating history, nor the same conditions (thermo-mechanic loads, integration of modifications, etc.). When the details are examined, this "de-standardization" increases when we shift from design to installation then to operation and it is this "de facto de-standardization" which prevents unfavorable situations from spreading throughout the entire park.

In practice and as experience has shown, the financial advantage provided by standardization is largely greater, by at least a decade, to the inconveniences associated with the generic defects. This is precisely what has lead the utilities of other countries to follow our standardization policy.

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LESSONS FROM PLANT MANAGEMENT FOR ACHIEVING ECONOMIC COMPETITIVENESS FOR EVOLUTIONARY REACTORS

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Abstract

Ontario Hydro, Canada's largest utility, has 20 nuclear power units of the CANDU design with a total design power of 15, 020 Electrical MW or about half of the total generating capacity. The performance of these plants, which were installed between the 1970s and early 1990s, was initially very good but worsened in recent years and the regulator expressed concerns about declining safety standards. In early 1997, an external Nuclear Performance Advisory Group led by Mr G. Carl Andognini was established and conducted an intensive assessment of the nuclear operation. That assessment ranked the operation of Ontario Hydro's plants as "minimally acceptable", noting a number of causes such as: lack of management leadership and accountability; poor safety culture; inadequate training; lack of configuration management; and, deficient organization. A recovery program is now underway, involving the laying-up of eight units to free resources to upgrade the remaining twelve. A number of lessons that can be drawn from the Ontario Hydro experience are presented. The key lessons include: the importance of having the right people, with the right qualifications in the right place, at the right time; the need for all staff to have a questioning attitude and to be committed to a "safety culture"; configuration management is essential - knowledge of the status of the plant must be maintained and all documentation kept up to date; maintenance must be given high priority and be provided adequate resources; defined standards are needed for the conduct of all work; the Directors (or equivalent) and senior management must understand the plant and appreciate the consequences of their decisions and actions. The paper concludes that good new designs are not sufficient by themselves to achieve economically competitive and safe nuclear plants - a high standard of operation and maintenance is also necessary throughout the life of the plants.

1. INTRODUCTION

This symposium focuses on the design of what has been termed "evolutionary" water cooled reactors, i.e., advanced designs of the pressurized water reactors (PWR), boiling water reactors (BWR), and pressurized heavy water reactors (PHWR's), which promise increased efficiency and safety, and improved economic competitiveness. In the context of the last point, economic competitiveness, the organizers have wisely included a session on "keys to economic viability". This paper contributes to that topic, "economic viability", by providing the important lessons that have been learned regarding the operation and management of Ontario Hydro's CANDU reactors.

The message is that while good design is essential to obtain economic competitiveness, it is not sufficient. To achieve the strong economic potential of an advanced nuclear power program, performance must be driven by excellence in nuclear operation and safety. This implies the need for a well planned maintenance program which is provided with adequate resources; a highly trained and motivated staff, and an overall sense of "safety culture" throughout the operating organization. It also implies the need for leadership by management in the interaction between technology, economics, human factors and safety, with adequate access to performance indicators to monitor performance.

It is the intent of this paper to outline the attributes of the management skills, processes and training that are needed to be implemented at the start up of any nuclear power plant to achieve high performance. This is done from the background of the Ontario Hydro experience, using the recovery program implemented at Ontario Hydro to restore performance as the reference program.

Table I Ontario Hydro Nuclear Stations

Station	Power	In service dates
Pickering A	4 x 542 MW	1971 - 1973
Bruce A	4 x 825 MW	1977 - 1979
Pickering B	4 x 540 MW	1983 - 1987
Bruce B	4 x 915 MW	1984 - 1987
Darlington	4 x 935 MW	1990 - 1993

This review emphasizes that safe, efficient and economic operation of nuclear power plants is an on-going challenge that will face changing requirements during plant operation and will require continuing efforts.

2. BACKGROUND

Ontario Hydro is Canada's largest electric utility. It was created in 1905 as provincially owned commission and is now a provincially owned corporation which has a monopoly for the generation, the setting of rates, and primary distribution of electrical power in the province. The provincial government has announced plans to open the electrical market to competition (as is happening throughout North America) and to change the structure of the corporation. The performance of Ontario Hydro's nuclear plants is essential if the nuclear option is retained in a competitive market.

Construction began on the first two commercial nuclear power plant units of the Pickering A station in the late 1960's. Subsequently, an intensive construction program was undertaken by Ontario Hydro leading to the current nuclear generating facilities in five stations as outlined in Table 1, comprising 20 units, with a total design capacity of 15,020 electrical MW.

The result of this intensive construction and generation program was the need for large numbers of operating and maintenance personnel to service the stations.

In 1996, Ontario Hydro had a total generation capacity of 29,800 megawatts, comprised of 6,500 MW hydraulic, 9,100 MW fossil-fuelled, and 14,200 MW nuclear. (The nuclear total was less than the design capacity primarily because Bruce A, unit 2, was removed from service due to significant steam generator problems.) In 1996 nuclear supplied almost 60 per cent of the electricity consumed in Ontario. In 1997, that contribution had fallen to less than 50 per cent, reflecting the falling capacity factors.

All of the Ontario Hydro nuclear power plants are of a CANDU design, basically similar to that of the CANDU 6 units at the Wolsong site here in Korea, which has been described in other papers at this symposium. A particular feature of CANDU is the ability to refuel on-power. This creates the potential to achieve capacity factors of 90 per cent or above. Capacity factors had been over 80 % in the early 1980s, were still about 70 % in 1995, but fell to less than 60 % the following year. The economic importance of this is illustrated by analysis showing that a deterioration of 20 percentage points in capacity factor is equivalent to a loss of in revenue of about $1.5 billion (Cdn) per year

In parallel with this decline in output, the regulator, the Atomic Energy Control Board (AECB), observed safety related problems and issued several warnings to Ontario Hydro. Licence periods, typically for two years in the Canadian system, were reduced to as short as six months in an effort to encourage the utility to make improvements. Several promises were made and plans were proposed but little real change occurred.

3. COMPREHENSIVE ASSESSMENT AND RECOVERY PLAN

In January 1997, Ontario Hydro's President commissioned G. Carl Andognini, Chief Nuclear Officer and several associates to conduct a "independent" assessment of Ontario Hydro Nuclear's operation. The Chief Nuclear Officer formed the Nuclear Performance Advisory Group (NPAG). which was chartered to:

- plan and implement an intrusive process of problem discovery and verification;
- document discovery activities and recommend corrective actions;
- develop an overall recovery plan and long term business plan;
- assist the Chief Nuclear Officer with implementation;
- return the performance of Ontario Hydro's nuclear plants to upper quartile of world nuclear plants by 2001;
- provide the Chief Nuclear Officer with a high level of confidence that achieved performance will be sustained and continue to improve.

In response to the first of the mandated tasks, the Nuclear Performance Advisory Group introduced and conducted the "Independent and Integrated Performance Assessment" (IIPA).

The IIPA was modelled on Procedure 93808 "Integrated Performance Assessment Process" of the United States Nuclear Regulatory Commission (USNRC). That process is a broad based assessment program that spans key departments and evaluates safety culture, quality of programs and procedures, general material condition, problem identification and problem resolution. A detailed implementation plan was developed with performance standards based largely on document INPO 97-002 "Performance Objectives and Criteria" issued by the Institute of Nuclear Power Operations (INPO).

The following performance areas were assessed: operations, maintenance, engineering, quality, radiation protection, chemistry, training, security, organizational effectiveness, emergency preparedness, regulatory affairs. The assessment, conducted between March and June 1997 involving 80 consultants as well as many utility staff, required more than 35,000 person-hours to complete.

In parallel with this overall assessment a detailed examination was conducted on several safety-related systems using a method modelled after USNRC Procedure 93801 "Safety System Functional Inspection". This involves a detailed and comprehensive examination into a system (or program) to determine its ability to fulfil the required safety function.

The final report of the Independent and Integrated Performance Assessment ranked the performance of Ontario Hydro Nuclear as "minimally acceptable" and listed the following basic causes of the decline in performance:

- general lack of managerial leadership and accountability
- poor safety culture and standards
- ineffective personnel performance and training
- non-standard and outdated processes and procedures
- degraded safe operating envelopes caused by unauthorized changes, lack of configuration management, poor monitoring and inspection and inadequate preventive maintenance

- deficient organizational functions, improper skill mix and shortages in key positions
- labour relations encumbrances that limit management effectiveness

A general observation was that the CANDU design is very robust, a feature that enabled the stations to continue to operate safely despite the less than adequate operations and maintenance programs that had prevailed for some time.

The results of the Independent and Integrated Performance Assessment were presented to the Board of Directors of the utility on August 12, 1997, with the warning that major and immediate action was needed. A Nuclear Asset Optimization Plan (NAOP) was proposed and endorsed by the Board of Directors.

The final decision from options provided to the Nuclear Performance Advisory Group was to operate and improve the newer stations (Bruce B, Pickering B and Darlington) and to lay-up, for eventual recovery, Pickering A and the three operating units at Bruce A; and to make available badly needed resources to perform the recovery of the Units (Unit 2 of Bruce A was already in a shutdown mode due to steam generator problems.). This concept, referred to as the 12 / 16 / 20 plan, foresees Pickering A being re-started in the period 2000 -2002, and the Bruce A plant in the period 2003 - 2009, provided a sound business case can be made for returning the units to service.

The Board of Directors of the utility endorsed the Nuclear Asset Optimization Plan as proposed and it is now underway.

As noted above in the findings of the Independent and Integrated Performance Assessment, the nature of union agreements was a significant potential impediment to the implementation of the Nuclear Asset Optimization Plan which included major transfers of personnel. After several months of intensive bargaining, agreements have been reached which enable the movement of large numbers of people from the laid up plants, especially Bruce A, to the other stations needing improvements.

To effect the Nuclear Asset Optimization Plan a detailed Integrated Improvement Plan (IIP) was developed. The IIP identifies 66 major projects to improve the performance of the 12 operating units. Each project has an assigned manager and a specific budget, the size of which varies considerably. "Configuration Management" and "Environmental Qualification" (of equipment) are the two largest projects. The total program has a budget of $1.6 billion (Cdn) over five years with most of the work scheduled to be completed prior to 2002.

In recognition of the importance and complexity of the Nuclear Asset Optimization Plan a senior vice-president position was created with the specific mandate to integrate the management of the NAOP with all of the other key functions and practices within Ontario Hydro as a whole.

The regulator, the Atomic Energy Control Board (AECB), is adding staff specifically to monitor the progress of the NAOP program. In this regard, frequent reports are made to the AECB, including appearances before the five member governing Board.

Implementing the Nuclear Asset Optimization Plan is an expensive and challenging process. In addition to the incremental costs associated with replacing the power previously supplied by the shutdown units at Pickering A and Bruce A, large sums have had to be invested in the restoration and upgrading of operating fossil plants. As well, several fossil units have been restarted from long term layup to replace the generation previously produced by the seven layed-up nuclear units. This could have been avoided through an on-going, well-planned and properly funded maintenance program throughout the life of the plants. For nuclear power to remain a significant component of the Ontario electrical sector in the coming competitive marketplace, such large restoration costs must be avoided in the future.

4. THE HUMAN ELEMENT

In common with other nuclear utilities experiencing declines in nuclear power plant performance, the principal finding of the Independent and Integrated Performance Assessment was that management had lost effectiveness and the morale of operating personnel had declined. There were, undoubtedly, a number of reasons for this. For over two decades, Ontario Hydro Nuclear (OHN) had a vigorous design and construction program that was the primary focus of the corporation and senior management. Plant operation was considered to be just a routine. Nevertheless, for the first 10 to 15 years of their lives the plants ran reasonably well, partially because of the sound CANDU design and partially because the operating staff was, generally, experienced. Operations personnel were, typically, the same people who had commissioned these and earlier CANDU plants and who knew them in detail.

When the construction phase ended (with the start-up of the Darlington units in the early 1990s), the management of Ontario Hydro Nuclear did not make the transition from design and construct to become focused on operations and maintenance.

Further, the Board of Directors, under pressure to keep rates low, repeatedly cut Ontario Hydro Nuclear's operational, maintenance, and administrative budgets. In 1993, they introduced a major reduction in staffing along with a de-centralization of many functions previously carried out at Head Office. Both of these decisions, which proved to be problematic, were noted in the report on the findings of the Independent and Integrated Performance Assessment. While the staff reductions were done "humanely" by offering generous severance packages and early retirement incentives, they led to the sudden departure of many key operating staff, and the utility lost much "corporate memory" and expertise.

The effect of the departures was compounded by the fact that the documentation of the plants had not been properly maintained. The experienced operators "knew" the plants but had not updated drawings or procedures. (This is the underlying reason for the current large project under the Integrated Improvement Plan for "configuration management".)

Another "human" factor was the nature of the union agreements that had led to excessively rigid division of one trade from another and restricted management's ability to move resources to where they were most needed. Considerable progress has been made over the past year in modifying these agreements.

5. NOT UNIQUE

As mentioned previously the problems described are NOT unique to Ontario Hydro, nor even to the nuclear industry. Similar situations have developed in many other nuclear utilities around the world, as may be determined by examining reports produced by Institute of Nuclear Power Operations and the World Association of Nuclear Owners (WANO). In the United States, the USNRC has required several nuclear utilities to undergo an assessment similar to our Independent and Integrated Performance Assessment. A number of utilities have been put on the regulators "watch list", meaning they were close to being shut down (and some were actually shut down).

This widespread deficiency in management of nuclear power plants has surfaced since the Three Mile Island accident in the USA in 1979, and the serious Chernobyl accident in the former USSR in 1986, both of which led to intensive reviews of the operation of nuclear plants and the introduction of the concept of "safety culture". (The International Nuclear Safety Advisory group of the IAEA defines "safety culture" as: *that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance.*)

An interesting study took place in the past summer of 1998, under the auspices of the IAEA, in the form of a two-day Working Group meeting to review the situation at Ontario Hydro Nuclear in comparison with that of several other utilities. While the focus was on safety and regulatory issues, the findings reflect each utilities' overall nuclear operating culture.

The three unit Millstone plant operated by Northeast Utilities and the Oskarshamn plant in Sweden were the other plants represented. The Working Group consisted of representatives of the utilities and of the regulators, together with some staff from the Agency.

To quote from the summary report of the Working Group:

Factors common to all plants were that during their early operation they belonged in the league of the best performing plants. They failed to adequately manage the transition from the design and construction stage into an era where the focus needed to be excellence in operation and maintenance.

A common feature is the fact that senior management failed to appreciate or recognize these symptoms or their significance and failed to take effective corrective action. A lack of an appropriate questioning attitude by senior management .... and the absence of an effective "corporate oversight" ... were contributing factors.

A particular challenge to the regulator ... is the difficulty in developing clear criteria for when regulatory action becomes necessary in the case of gradually declining safety culture and associated management performance.

Due to the extreme pressures placed on the utility and regulator during recovery, the regulator has to develop new processes to effectively deal with the complex technical, human and public confidence issues. This takes considerable resources and new skills to manage. The regulator therefore has to look at invoking similar management and process changes as the utility to enable successful and sustaining response to the situation.

6. LESSONS TO BE LEARNED

Although the "evolutionary" plants being discussed at this Symposium will have enhanced safety features and improved quality, they will still need to be operated and maintained to high standards if they are to provide the high capacity factors and long life necessary to be competitive and to pay back the large investment required for their construction.

From the experience at Ontario Hydro, especially the results of the IIPA process, it is possible to list a number of issues that must be addressed and programs put in place to achieve successful operation of nuclear power plants. These can be grouped in various ways - this paper will use "people", "plant", "process", and "organization.

6.1 People

People are the most important asset of a utility operating a nuclear power plant. It is essential to have the right people with the right qualifications, in the right numbers, in the right place, at the right time. There must be clear definitions of roles, responsibilities, authority, and accountability, and the establishment of a "requisite management" approach. Requisite Management specifically addresses organizational hierarchy and accountabilities. There should be measurable performance criteria.

All staff, at all levels, must have a sense of, and be committed to, a "safety culture", which includes having a questioning attitude.

An appropriate working environment must be provided, including facilities, qualifications, equipment, tools and procedures. Each individual staff member must be adequately trained for his or her specific role.

There is a special need to develop and maintain the managerial capability of all levels of managers and supervisors. Everyone must be held accountable for their individual outcomes.

6.2 Plant

It is important to maintain knowledge of, and control of, the status of all plant equipment. This helps ensure that the level of "backlogs" remain acceptable, such as: corrective maintenance; disabled alarms; control room instrumentation out-of-service; long term equipment deficiencies; temporary leak repairs and jumpers.

Maintenance must be given high priority and a maintenance strategy articulated. There is a need for appropriate preventative and predictive maintenance programs to augment non-routine (or corrective) maintenance, and all maintenance programs must be provided with adequate resources (qualified personnel and funding).

6.3 Processes

There needs to be clearly defined standards and management expectations for: conduct of operation, conduct of maintenance and conduct of engineering

High and low level performance monitoring systems, with identifiable and measurable criteria, are needed to provide early warning of any departure from acceptable levels of performance. Managed systems should include performance indicators and goals for every level of the organization from the Boardroom to the shop floor.

It is essential that the documentation of the plant be kept up to date to reflect the plant as it exists. This includes drawings, design documents, safety and other analyses, equipment manuals, and any other documents describing the plant. (Departures from known configuration can, and have, led to encroachment of the design safety margins for systems critical to nuclear safety; such as, unauthorized material substitutions; unauthorized and unreviewed alterations.)

Operating and maintenance manuals need to be reviewed and updated frequently to reflect the plant as it exists, and to take into account operating experience and equipment behaviour.

All aspects of the operation and maintenance of the nuclear power plant must meet high standards such as those issued by the INPO/WANO in its documents on: objectives and criteria; guidelines; and, good practices.

Engineering should be assigned an appropriate and visible role in the overall operation of a plant.

Outages need to be well managed; implying being planned in advance and having appropriate resources (qualified personnel and equipment) made available.

6.4 Organization

It is essential that the Board of Directors (or equivalent body) and senior management understand the operation of the plant(s) within their mandate and the consequences of their decisions and actions; i.e., senior executives must have the business acumen to provide the leadership necessary to manage the interaction between technology, economics, human factors and safety, and to have available performance indicators to detect early symptoms of shortcomings and declines in performance. (At Ontario Hydro a "Nuclear Review Committee" and "Nuclear Oversight Committee" have been formed to ensure that the Board of Directors and senior management are fully informed on issues associated with the operation of the nuclear plants.)

In large multi-station utilities, it is highly desirable that organizational structures, practices, and procedures should be consistent throughout the organization.

7. CONCLUSION

While desirable and probably necessary if nuclear power is to remain competitive, new designs, by themselves, are not sufficient for a successful nuclear power program. Nuclear plants must be operated with great care, attention to detail, and with an appropriate emphasis on safety. If sufficient attention and resources are not dedicated to the operation, the potential of the new plants will not be realized. The experience at Ontario Hydro and other nuclear utilities is that if the stage of declining performance is reached, recovery usually requires an expensive program to remedy deficiencies including replacement of the management structure.

DESIGN MEASURES IN EVOLUTIONARY WATER COOLED REACTORS TO OPTIMIZE FOR ECONOMIC VIABILITY

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Abstract

Since the mid 1980s, there have been various efforts to develop evolutionary water cooled reactors based on the current operating plant experience. To sustain and improve the economic viability, particular attention has been paid to the following aspects in developing evolutionary water cooled reactors: design simplification and increased operating margins, standardization in design as well as construction and operation, integration of operating plant insights, and consideration of safety, operability and constructibility during the design stage. This paper reviews each item and discusses several examples from some of the evolutionary water cooled reactors being developed.

1. INTRODUCTION

Since the TMI-2 incident, the added regulatory burden and complexity to the plants have decreased the economic competitiveness of nuclear power. To remedy the situation, nuclear utilities worldwide have examined the requirement for the future light water reactors. In the USA, the Electric Power Research Institute developed the ALWR Utility Requirements Document [1]. European countries have developed European Utility Requirements for ALWR [2]. In Korea, KEPCO is currently developing the ALWR requirement for Korean utilities. AECL has worked closely with CANDU utilities to establish improvement requirements for the evolutionary development of the heavy water reactors [3]. All these efforts recognize the importance of economic viability and have proposed:

- Simplification and increased margins
- Standard design with repeated construction
- Integration of operating plant insights and
- the consideration of safety, operability and constructibility during the design stage.

The recent trend of deregulation of electricity market emphasizes further plant economy. In a deregulated environment, the large capital cost of nuclear compared to alternate energy sources could be a handicap if not compensated with a substantial decrease in generation costs. Shorter construction schedules, simplification of designs together with regulatory stability are thus becoming increasingly important to control costs.

Advanced water-cooled reactor designs have progressed worldwide. Examples are ABWR, AP600, CANDU9, EPR, KNGR, and System 80+ among others. ABWR (Advanced Boiling Water Reactors) is an evolutionary boiling water reactor developed by GE in cooperation with Hitachi and Toshiba. AP600 is a passive PWR developed by Westinghouse in USA. CANDU9 is a 900 MWe evolutionary HWR plant developed by AECL, Canada. EPR is an evolutionary PWR being developed by NPI. This is a joint effort between France and Germany. KNGR is an evolutionary PWR being developed by KEPCO, Korea. System 80+ is an evolutionary PWR developed by ABB-CE, USA. All of these plants were developed with the consideration of the utility requirements. Detailed descriptions of advanced light water reactor developments can be found in Ref. [4].

In this paper, we review the design improvement consideration for economic viability. More specifically, we will examine the design improvement for the four points described above with some examples.

2. SIMPLIFICATION AND INCREASED MARGIN

2.1. Simplification

We have observed the substantial increase in complexity of nuclear power plants. Stahlkopf & Chapin [5] have compared the U.S. nuclear plants built in late 60s with those built in late 70s. As a measure, concrete quantity increased from 90 cubic yd/MWe to 248 cubic yd/MWe. Cable increased from 2000 ft/MWe to 4889 Mwe. These are about 280% increase. Furthermore, the increase in craft labor was staggering: from 3500 person-hr/MWe to 21600 person-hr/MWe. One significant contributor is the substantial increase in regulatory requirements on safety systems. However, in addition to the regulatory complexity, there has been a lack of attention to adequate simplification on the part of designers over the years. The plant modification has been done too commonly by an ad-hoc process from the existing design.

The increase in complexity could result, if not adequately controlled, in the increase of the maintenance and construction cost as well as in the difficulty of operating the plants. Adding a component to the plant that is inexpensive in itself could become an expensive proposition from the life cycle point of view. As an example, Figure 1 shows the effect of removing a pump in a US environment in mid 1980s, illustrated by W. Sugnet [6].

The effort by ALWR developers on simplification can be summarized by the ALWR simplification requirement [7]:

- Use a minimum number of systems, valves, pumps, instruments and other mechanical and electrical equipment
- Provide a man-machine interface which will simplify plant operation and reflect operator needs and capabilities [8]
- Provide system and component designs which assure that plant evolution minimizes demands on the operator
- Design equipment and arrangements which simplify and facilitate maintenance;
- Provide simplified protective logic and actuation systems, and provide automation to reduce operator testing tasks
- Use standardized components to facilitate operation and maintenance

Below, we will review several examples on simplification from the current evolutionary water cooled reactor development.

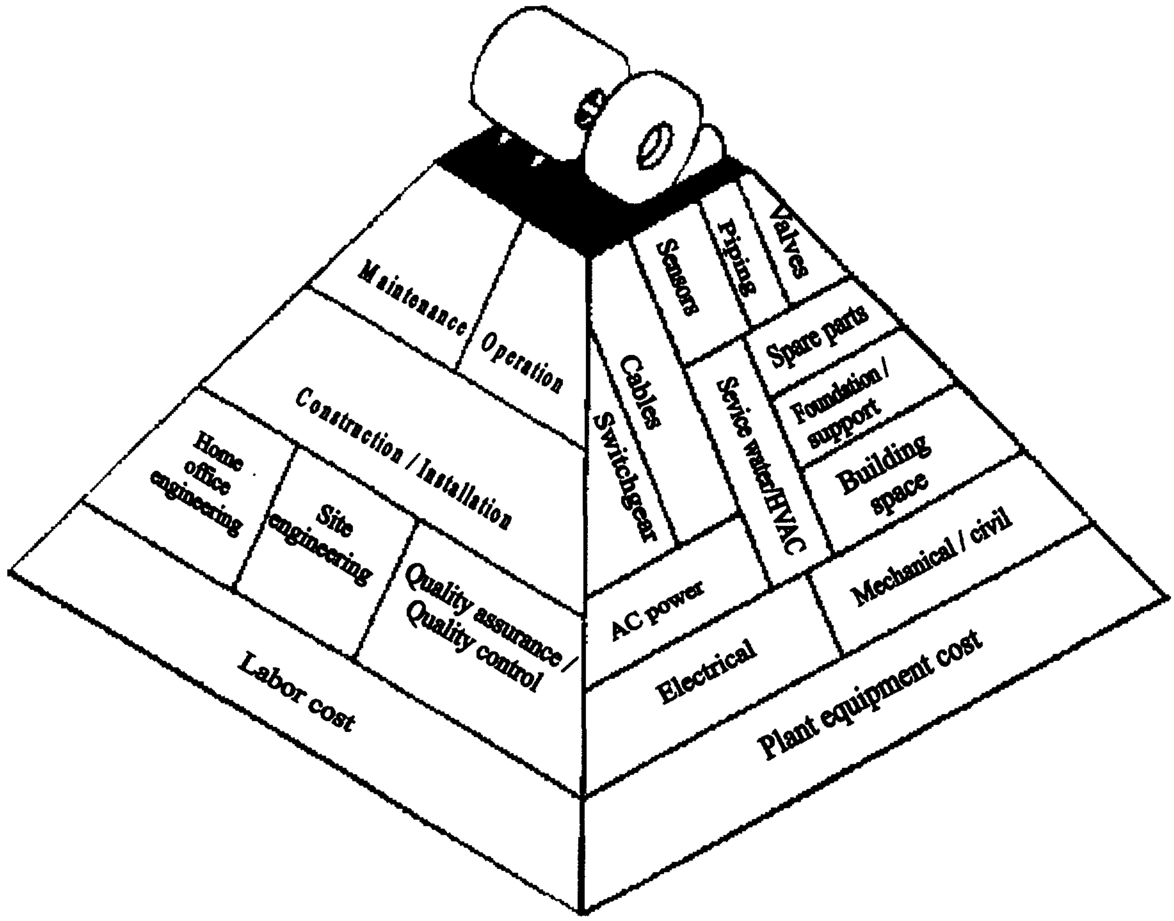

FIG. 1 Removal of active components has great potential for simplification.

Most of the evolutionary designs have adopted four-train configuration for important safety systems and their support systems to increase the safety of the plant. This, in itself, could increase the complexity. However, in KNGR as well as System 80+, the optimization has been achieved by removing low-pressure safety injection while increasing the capability of safety injection pump and safety injection tank. Furthermore, the injection is routed directly into the reactor vessel. Previously, the injection is through the cold leg. Direct vessel injection raises a concern on possible pressurized thermal shock of reactor vessel. Ring-forged reactor pressure vessel is used to alleviate pressurized thermal shock issue. This has an additional advantage by reducing the number of welds and corresponding inspection activities. The long-term decay removal is handled by shutdown cooling system.

For the EPR, important safety systems and their support functions are arranged in a four-train configuration. Examples are safety injection, emergency feedwater, component cooling, and emergency electric power system. The high degree of redundancy has significant advantages regarding an optimized preventive maintenance concept that makes possible maintenance and inspection during operation. Thus, plant outage time is reduced and plant availability increased.

The design of the CANDU 9 emergency core cooling (ECC) system has been simplified by reducing the number valves and using passive on-way rupture disks to separate the reactor cooling system from the emergency core cooling system. In addition, the ECC water tanks for high pressure injection were located inside containment with shorter injection lines directly into the reactor headers instead of the heat transport system piping. These simplifications will increase the reliability of this system over that of previous designs which reduce the capital cost as well as significantly reduce the operating and maintenance costs for testing, inspection, maintenance and repair over the lifetime of the plant.

KNGR has simplified reactor vessel head area by utilizing an integrated reactor vessel head package. An Integrated Head Assembly (IHA) is developed to incorporate all of the reactor vessel head components into one module. The IHA casing is designed so that one can use the multiple stud tensioner during refueling outage. The multiple stud tensioner allows simultaneously detensioning and tensioning of all the reactor vessel studs. This enables the removal of the head area components and reactor vessel head at once. The use of IHA is estimated to save almost three days in comparison to typical seven-day schedule for the reactor vessel head disassembly.

Most of the evolutionary water-cooled reactor designs are equipped with an advanced digital I&C system. This enables self-testing of the system which reduces the effort required for maintenance and testing. In conjunction with the digital system, the use of multiplexing or data-highway simplifies cable routing and reduces the amount of cables and connections.

2.2. Increased margin

Nuclear power plants have operated as a base load generating facility in most cases. For the base load operation, the sensitivity study on the cost of electricity indicates that the plant availability is one of the most important parameters. The increase in the operating margin may increase the initial cost but it improves the plant availability greatly by reducing the chance of shutdown. The increase of the margin has been one of common requirements among EPRI URD, EUR, and K-URD.

To improve the operating margins, evolutionary PWR and PHWR designs incorporate larger pressurizer volume and steam generator inventory. For example, System 80+ designed by ABB/CE has 33% larger pressurizer volume and 25% larger steam generator secondary inventory compared to System 80 design [4]. The increase of the steam generator secondary inventory relaxes the operator action time in the event of loss of feedwater transient. In CANDU 9, the larger pressurizer volume is capable of accomodating changes in reactor coolant volume from 100 °C to full power, which enhances the natural circulation capability during system cooldown after a loss of flow. Most of the evolutionary plants are designed with the goal of minimum operator action time of 30 minutes. As an example of design change, an orifice has been integrated to the letdown line in KNGR so that operator has more than 30 minutes to close the isolation valve.

These simplifications and improved operating margins yield economic benefits such as high availability, outage reduction, reduced staffing level, and spare parts reduction.

3. STANDARD DESIGN WITH REPEATED CONSTRUCTION

The use of the standard design yields the benefit of scale, learning effect, and licensing stability. The use of several or more standard plants provides, by sharing of resources, the benefit in training, maintenance, spare parts, and procurement. It will reduce the cost both in construction and operation. These benefits have been experienced by Electricite de France (EdF) as well as the Westinghouse SNUPPS family and Palo Verde plants in USA. The replication of the CANDU 6 plants from Canada to Argentina and Korea and subsequent additions of 3 more CANDU 6 units at Wolsong site have yielded progressively improved construction schedule and costs. In Korea, the KSNP series of PWRs (YGN3&4, UCN3&4) shows the similar advantage. Most, if not all, of ALWR being developed pursue the standard plant concept.

In USA, the standardization has become a part of the design certification rule. Operating plants in USA have been licensed on a plant-specific basis. It is a two-step process, approval of PSAR and

FSAR with the license. One-step licensing process has been legislated for ALWR through 10 CFR 52. Subpart B of 10 CFR 52 covers the standard design certification. Design once certified under 10 CFR 52 rule will be protected from new requirements or public hearings except under special circumstances. This process would eliminate the licensing instability that has been the mode of operation since TMI-2. Currently, two evolutionary LWRs, ABWR and System 80+, have been certified as a standard design in U.S. which will remain effective for 15 years. A passive PWR, AP600, is currently in the process of design certification. Similar licensing process is being pursued in Korea. KNGR design is planned to be submitted in 1999 for the review under the standard design certification rule. With the requirement that the CANDU 9 design should be licensable for both domestic and foreign potential users, the pre-project Basic Design Engineering program included a two-year formal extensive review by the Canadian regulatory Agency. The same licensing review process in the granting of the construction license for a nuclear project in Canada was used. The licensing review has been successfully completed and the final report from the AECB was issued in January 1997 [9].

The benefit of standardization can be maximized by early and broad utility involvement and a life cycle commitment by utilities. Utility participation early on the design phase was common in most of the evolutionary plant development. In USA, Utility Steering Committee was formed under EPRI ALWR program from the beginning to review and feedback the ALWR development effort. EdF and German utilities are closely working with NPI on EPR development. In Korea, KEPCO, who is the electric utility in Korea, manages the development of KNGR. Commercial standardization has been an important element of "first-of-a-kind engineering" performed by U.S. vendors on ABWR and AP-600. In this stage, equipment purchase specifications for major equipment are prepared. It is planned that these specifications will be used repeatedly. Purchase specifications of other equipment that are important to the standardization will be developed on a form-fit-function basis [10]. Commercial standardization requires broad consensus among participants for the design of the first plant and duplication of the first plant design details as well as lessons learned from construction,startup testing and operating experience. In this regard, in USA, Nuclear Power Oversight Committee (NPOC) made a policy statement.

"Nuclear power plant standardization is a life-cycle commitment to the uniformity in the design, construction and operation of a family of nuclear power plants. Rigorous implementation of standardization is expected to achieve the efficiency and economy typically associated with increases in scale or breakthroughs in technology." [10]

In developing EPR, the standardization concept has been applied not only between the plants but also within each plant. From the beginning of the EPR basic design phase, intensive engineering work was dedicated to compiling catalogues of equipment such as embedded parts, supports, piping, valves, and pumps, etc. Identical components, such as pumps and heat exchangers will be used as much as possible to simplify maintenance. In addition, utilities and vendors joined force to set up common codes related to the EPR design, "EPR Technical Codes". These codes will promote standardization of quality requirements and competitiveness through the use of industrial standards.

It has been shown that the repeated construction of the standard design saves the construction cost greatly. Based on the estimation made by EdF [11], the repeated construction reduces capital cost ratio per kW by 25 to 30 % by the second unit. Further repeated construction would reduce this by 40% based on the first unit cost. KEPCO has experienced similar reduction in equipment and engineering service expenses.

To benefit further from the standard design and to reduce the construction schedule, the modular construction approach has been adopted in varying degrees by all water-cooled reactor developers. Modularization has been used for some time in the area such as water treatment, demineralizer package, air compressors, and T/G subassemblies. This has been extended to structural modules as well as wider range of equipment modules. ABWR has utilized the modular construction approach. This has been shown to be essential to meet the EPRI URD required schedule of 48 month from first concrete to fuel loading. In CANDU 9, special features related to constructability are incorporated, which will result in the reduction of the construction costs and schedule. The CANDU 9 design includes a site layout and building arrangement that promotes efficient construction and incorporates the increased use of prefabrication - ranging from skid mounted packages to large structural assemblies. In Korea, KNGR design team is developing modules for the Aux. Building below grade level to improve the construction schedule. Modularization decision is made in tandem with the development of the construction schedule to maximize its benefit.

4. INTEGRATION OF OPERATING PLANT INSIGHTS

Evolutionary reactor designs are based on operating plants and improve the design based on the operating experience. For example, System 80+ is improved from System 80. Three System 80 plants have been in operation at Palo Verde site in USA. KNGR is based on operating 1000 MWe KSNP design. Three KSNP units (YGN 3&4, UCN3) are in operation. By having the operating plants as a basis, the operating plant insights have been reflected in designing the evolutionary plants. Furthermore, utility requirements developed by EPRI, by European utilities and others essentially embodies the operating experience and desired improvements. The utility requirements have been the backbone of all the evolutionary water reactor development. Attention to the operating plant experience and insights assures that the new design will be safer and economical to operate when being built.

The reduction of the radiation worker exposure is important not just for the ALARA but also for the operating cost. The KNGR design team has sought to incorporate those lessons learned by the current generation of nuclear power plant to meet the exposure limit of 20 mSv/yr set by ICRP60 and to limit the collective exposure to less than 1 person-Sv/yr which is one of the EPRI URD goals. Radiation exposure from the maintenance work of KEPCO nuclear plants has been compiled and high exposure maintenance activities have been identified. Not surprisingly, the highest exposure maintenance work is the steam generator tube inspection. The improvement in the access to steam generators and the use of Inconel 690 for SG tubing are some of the consideration given to reduce occupational radiation exposure. Other means of achieving the exposure goals are summarized as follows:

(1) The generation of crud is controlled by adopting the material with low cobalt impurities, and maintaining the pH of reactor coolant water in the range of 6.9 to 7.4.

(2) Wider use of ion exchangers instead of evaporators in radwaste system.

(3) Use of permanent and temporary shielding as an integral part of KNGR design.

5. CONSIDERATION OF SAFETY, OPERABILITY AND CONSTRUCTIBILITY DURING DESIGN STAGE

Unlike previous nuclear plant design, the aspect of safety, operability, constructability, and operating plant insight has been systematically considered from the beginning of the design and during the design process. For example, both ABB/CE and GE performed the PSA during the design stage and reflected in their design improvement [12]. Furthermore, ABWR has performed constructability review, construction schedule development as well as cost estimate as the part of the first-of-kind-engineering program. In the CANDU 9 design program, CANDU utilities' operating staff participated in formal design reviews and joint constructability studies were performed with construction companies to maximize the benefits of past experience, resulting in cost reduction during future plant construction and operation. It is our understanding that similar activities are being performed in AP600, as well as EPR program.

As a part of the KNGR design development, KEPCO systematically evaluates the design periodically on safety, constructability, operability, ORE, and cost impact. The result of the evaluation is being used to optimize the design further. As an example, probabilistic safety analysis result showed that the auxiliary feedwater system reliabilitycan be improved by changing the valve arrangement and the removal of a valve. Also, automatic operation of the start-up feedwater pump was recommended based on the PSA study [13].

Cost estimate during the design stage helps to optimize the ALWR development. Latest generation cost estimate of EPR shows that it is at least 10% lower than the cost prediction for future coal plants as well as that for combined cycle gas turbines. With the current power rating, investment costs are slightly below that for N4, in terms of $/kWe. Availability improvement together with reduced construction time among others, allow further substantial decrease in terms of generation cost. Latest cost estimate of KNGR shows also that it has similar cost advantage over the coal plants.

6. CONCLUSION

In this paper, we reviewed the effort taken into making the evolutionary water-cooled reactors economically viable. We reviewed some of the examples in the area of simplification, standardization, and reflection of operating plant insights. In general, the evolutionary water-cooled reactors do have higher investment cost due to more stringent safety requirement. However, the extra costs are counterbalanced by performance advances in plant availability and shorter construction schedule and attention to the cost from the beginning of the design phase. For example, the recent OECD study shows the cost competitiveness of a 1300 MWe ALWR plant against other electricity generating facilities. The effort of the simplification, standardization, and the reflection of operating plant insights in developing evolutionary water-cooled reactors do pay off.

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POSITIVE EXPERIENCE IN THE CONSTRUCTION AND PROJECT MANAGEMENT OF KASHIWAZAKI-KARIWA #6 AND #7

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Abstract

Kashiwazaki-Kariwa #6 and #7 (K-6/7), the world first Advanced Boiling Water Reactor (ABWR) units, started commercial operations on November 7, 1996 and July 2, 1997 respectively. ABWR has been developed as a standard design of the next generation BWR to meet common goals set by the Japanese electric utilities and BWR manufacturers (GE, Hitachi and Toshiba) based on our design, construction, operation and maintenance experience of nuclear power plants. The construction of K-6/7 and confirmatory tests for the verification of the first-of-a-kind (FOAK) design features of ABWR were conducted smoothly without any delay. The duration of the construction was 51.5 months. It was shorter than conventional BWRs in Japan by nearly one year. This was realized by design features of ABWR for better constructability, a principle of "test before use" applied to the FOAKs, advanced construction technology, detailed engineering at very early stages of the project, and good construction management. The positive experience in the K-6/7 project is now being reviewed and standardized for next ABWR projects. The data and knowledge accumulated through the K-6/7 project will be utilized effectively with the aide of the latest information technology.

1. INTRODUCTION

Kashiwazaki-Kariwa #6 and #7 (K-6/7), the first Advanced Boiling Water Reactor (ABWR) units, started full commercial operations on November 7, 1996 and July 2, 1997 respectively. Since the introduction of our first reactor in 1971, TEPCO, in collaboration with other Japanese electric utilities and plant manufacturers, has been addressing various issues of nuclear power plants to improve their safety, reliability, operability, occupational radiation exposure, economy, and so on. There were two major programs launched in 1970s for that purpose: One was to improve and standardize the technology originally developed by GE. The other was to develop a state-of-the-art plant design as a standard for next generation BWRs, and this program gave birth to ABWR.

ABWR is the first reactor that has been developed to meet common goals set by the Japanese electric utilities and BWR manufacturers (GE, Hitachi and Toshiba), based on our design, construction, operation and maintenance experience of nuclear power plants. Its development goals were:

(1) Better safety and reliability.

(2) Reduction in occupational radiation exposure and in the amount of radioactive waste.

(3) Better operability and maintainability.

(4) Better economy.

In order to achieve the development goals, ABWR has incorporated various first-of-a-kind (FOAK) equipment and systems. These are Reactor Internal Pump (RIP), Fine Motion Control Rod Drive (FMCRD), Reinforced Concrete Containment Vessel (RCCV), three-division Emergency Core Cooling System, high efficiency BOP system with a large capacity turbine generator, full digital control system with advanced man-machine interface, and so forth.

This paper reviews how the K-6/7 project has been successfully completed fulfilling every development target even though many FOAK design features are included in its design.

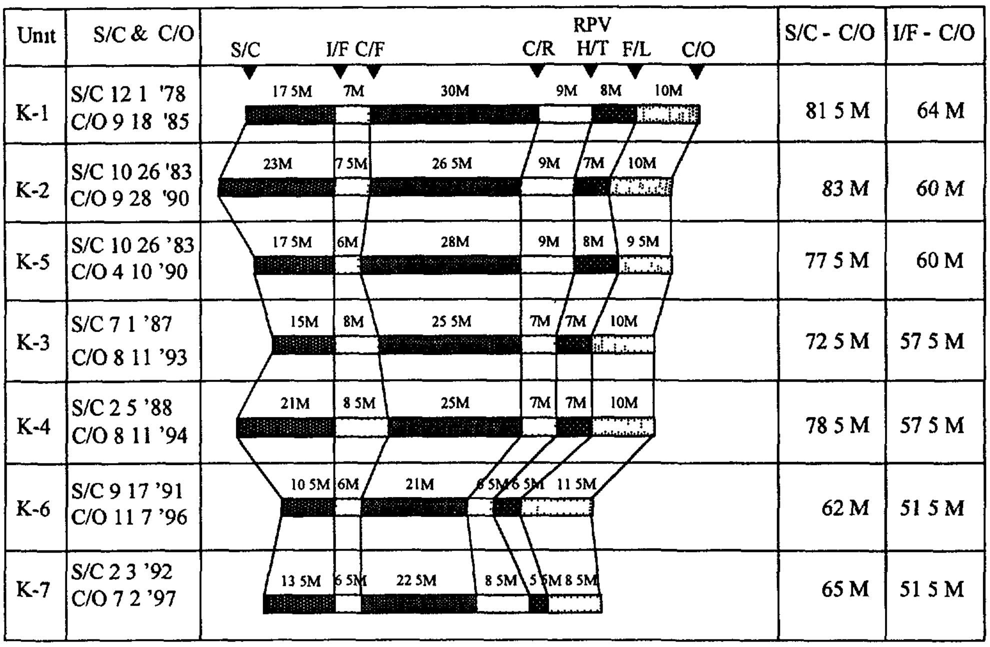

FIG. 1. Construction duration of Kashiwazaki-Kariwa Nuclear Power Station.
(S/C· start of construction, I/F· inspection of foundation, C/F completion of foundation mat, C/R completion of refueling floor, RPV H/T RPV first hydrostatic test, F/L· fuel loading, C/O. start of commercial operation)

2. CONSTRUCTION DURATION

Kashiwazaki-Kariwa is located in a district facing the Sea of Japan, an area known for strong seasonal winds and heavy snowfalls in winter. On account of that, there are limitations on construction activities such as unloading heavy cargo at the wharf in winter. The bedrock that supports principal buildings is at a depth of 25 to 40 meters underground. The reactor building (R/B), therefore, is a semi-underground type structure, which requires a long excavation period before starting the building construction. The reduction in the construction period has long been a challenging target under these difficult conditions.

Figure 1 shows the construction periods of K-6/7 and other BWR units of Kashiwazaki-Kariwa

The time required before the inspection of the foundation was devoted to the excavation work, therefore it depended on the ground conditions. After the inspection, actual construction of the buildings started. For the purpose of comparison, the term "construction period," or "construction duration" refers to the time from the inspection of the foundation to the start of commercial operation in this paper[1].

In the 1980s and the first half of the 1990s, there was a strong demand in Japan to reduce construction period because of a fast growing electricity demand and high interest rates. It was also a period we experienced changing labor market conditions such as a shortage of skilled field workers

[1] It may be noted that the normal definition of the construction period refers to "from pouring of first structural concrete to start commercial operation".

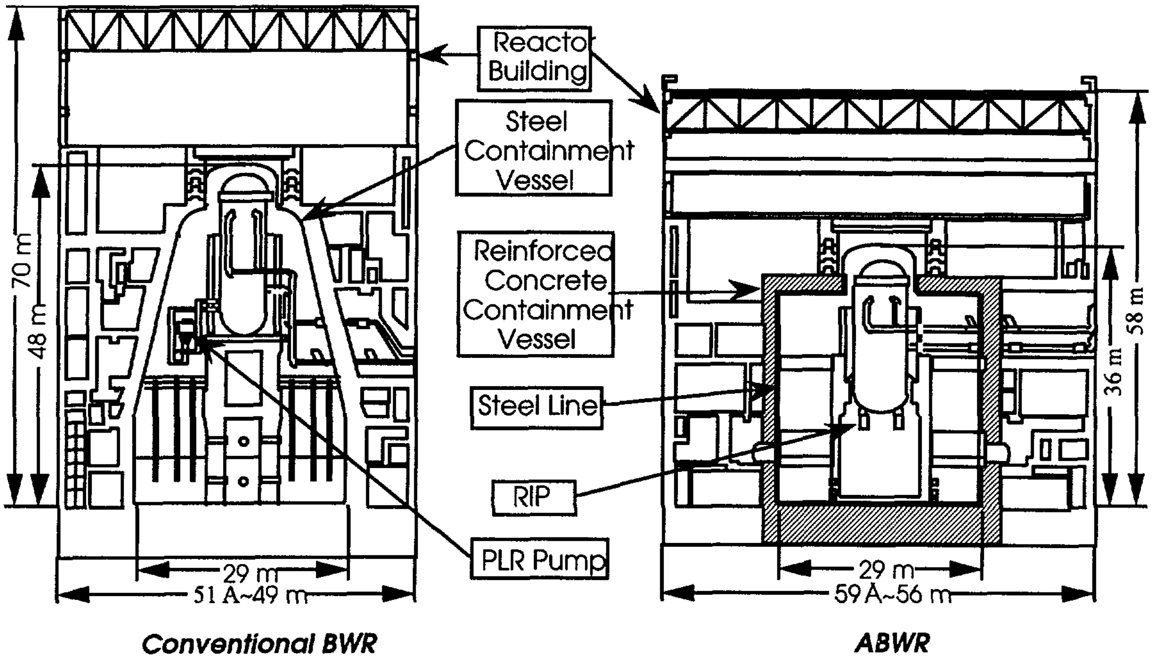

FIG. 2. Reinforced Concrete Containment Vessel (RCCV) and Reactor Building.

and steep rises in labor wages. These are the reasons why TEPCO has been making every effort to improve the productivity in the construction of power plants.

The construction duration of K-6 was 51.5 months, and the same length of the schedule was required for K-7. It was sharply improved from other conventional BWR units. If some conditions unique to K-6/7 such as additional tests for the FOAKs had not been included in the schedule, it could have been about 48 months. In addition, it has a potential for further reduction depending on the labor markets. The schedule from the inspection of the foundation to the start of the fuel loading is determined by civil and architectural work. In the case of K-6/7, its duration was about 40 months with regular holidays of 6 days per month. A total of 24 days of additional holidays per year were also given on such occasions as the new year. We can shorten the duration of 40 months to about 35 months in next projects by changing the holiday schedule to be 4 days per month.

The improvement in the duration of the plant construction was realized by:

(1) Design features of ABWR for better plant constructability.

(2) Prudent design change control with a principle of "test before use" applied to the FOAKs.

(3) Advanced construction technology.

(4) Detailed engineering at very early stages of the project.

(5) Good construction management.

3. DESIGN IMPROVEMENT FOR SHORT CONSTRUCTION PERIOD

ABWR has many new technical features. Those features which have a significant bearing on the construction scheme include the use of the RCCV structurally integrated with R/B and a reduction in the building volume through the rationalization of the layout design.

In the scheme of conventional BWR construction, installation of the steel containment vessel holds a part of the critical schedule. Only after the installation and pressure tests on the containment vessel are completed, the R/B structure surrounding it can be erected.

On the other hand, the RCCV is integrated with the R/B as shown in Figure 2, therefore, its installation can proceed concurrently with the erection of the R/B. This provides a merit of shorter construction duration. The time required to install its liner plates can be offset by adopting a large block construction method.

The building volume has been significantly reduced in ABWR. The adoption of low-NPSH RIPs, which have replaced large Primary Loop Recirculation Pumps located under the Reactor Pressure Vessel of conventional BWR, has enabled us to reduce the space in the R/B. Unlike self-standing steel containment vessels, the RCCV is integrated into the R/B structure, therefore, it has more flexibility in the design of its shape. The layout design has been optimized by making the best use of these advantages.

As a result, the total building volume of K-6/7 is smaller than that of K-3/4, the latest BWR-5 type units, by more than 20% even though the plant power generation capacity has been increased by more than 20%. This has led to the reduction in building material, piping, cables, ducts, and other plant utilities, which also has contributed to short installation duration.

4. DESIGN CHANGE CONTROL

K-6/7 are the first ABWR units, and a lot of FOAK equipment and systems have been adopted in the design. One of the priorities placed on this project was the design change control. The project had not been completed as scheduled without good management over the FOAK design. Our past experience shows that many troubles occurring in new plants were related to design changes. In order to minimize the risks associated with design changes, every small deviation from conventional design should be picked out, and then, it must be carefully evaluated from the view points of performance, reliability, transient behavior, side effects on other systems, and so on. TEPCO has established a rule of design change control through a series of continuous construction of nuclear power plants, and has successfully reduced the occurrence of teething troubles with new units.

4.1 Design review

As a part of the design change control, TEPCO's in-house design reviews by Design Review Committee and extensive design reviews called "Juten Sekkei Review" by joint task forces of TEPCO and plant suppliers were regularly conducted during basic and detailed engineering phases of the K-6/7 project.

TABLE I. LIST OF JOINT EXTENSIVE DESIGN REVIEWS, "JUTEN SEKKEI REVIEW"

First of a kind mechanical system	**First of a kind electrical system**
Reinforced Concrete Containment Vessel	RIP Adjustable Speed Driver
Main Steam & Feedwater Piping in RCCV	CRD Stepping Motor
Lower Drywell Arrangement	Main Control Panel
Reactor Pressure Vessel	RC & IS
Reactor Internals	SSLC
New Type Hf Control Rod	PRNM
Reactor Internal Pump	Digital Control System
Fine Motion Control Rod Drive	Excitation System
HCU	**Scaled up equipment**
RCIC (first application to ECCS)	MSIV
Moisture Separator Reheater	Main Steam Turbine
Heater Drain Pump Forward System	Main Condenser
Off-gas System	Main Generator
Cross-around Piping	**Others**
	Water Chemistry

For K-6/7, Juten Sekkei Reviews were conducted on 27 areas, as shown in Table I, including the FOAK equipment such as RIPs and scaled-up equipment such as the main condenser. Taking an example of the RIPs, the pumps and their motors were reviewed from the viewpoints of functional requirements, component design details, operational reliability, maintainability, and consistency with the licensing matters.

The new design features and the scaled-up equipment were also reviewed from the viewpoint of manufacturability. Material qualification tests and production mock-up tests were conducted on some areas as a part of design reviews.

Discussions had also been made throughout the construction period to determine whether the units had a possibility to cause similar troubles happened at the other existing nuclear power plants. For example, after TEPCO learned droplet erosion on the surface of the condenser titanium tubes in foreign power plants, we reviewed our maintenance records of operating units and found a symptom. Then, discussions were held with manufacturers and the provisions were made in the design of the K-6/7 main condensers to reduce the speed of condensed droplets attacking on the tubes. In addition, we made it a practice that all the available trouble records around the world were revisited before the start of commercial operation. It was a final check-up to ensure that every experience had been reflected in the units.

When the review was finished, a design change control sheet was filled out for each component or system by relevant engineering sections at TEPCO site office. A summary of the review results and verification test plans were documented in the sheet, and it was circulated to the section in charge of the verification tests.

4.2 Design verification test

There were many large-scale tests conducted in joint studies of the Japanese electric utilities and manufacturers for the verification of the new design features. A principle of "test before use" was set and strictly applied to the FOAKs. In the case of the RIP, a series of full-scale 1/10-sector model tests confirmed its performance, control characteristics, seismic capability, and transient behavior. Flow-induced vibration of the reactor internals was evaluated from the results of 1/5-scale full-sector tests and full-scale 1/10-sector tests. In addition to the joint studies in private sectors, a series of full-size design verification tests were also conducted by the government authority at a 1/5-sector test facility with two RIPs installed side by side.

4.3 Product verification test

The product verification tests were conducted at shops on a component basis and at the site on a system basis. When each test was completed, its results were evaluated and documented, and their summaries were recorded in the design change control sheet. It was then circulated back to engineering sections for their final confirmation.

In the case of the RIP, special tests were added to ordinary pump tests. Those included were a confirmation test of its coast down characteristic, a run back test, unbalanced operations of 10 RIPs, a restarting test under reverse flow conditions, and so forth.

5. ADVANCED CONSTRUCTION TECHNOLOGY

Construction methods have been improved based on the experience accumulated through the continuous construction of nuclear power plants. Table II shows principal ideas of the advanced construction methods that have been used for the construction of Kashiwazaki-Kariwa Nuclear Power Station. These methods were adopted not only for the reduction in the construction duration, but also for enhanced quality, assurance of on-the-job-safety, and reduction in the construction cost. There are two key methods that determine the outline of the construction schemes. One is called "All-weather

TABLE II. VARIOUS CONSTRUCTION METHODS INTRODUCED TO KASHIWAZAKI-KARIWA

Objective	Construction method	Applied unit #1	#2	#3	#4	#5	#6	#7
Concurrent progress in parallel construction	Large block construction	✖	✖	✖	●	○	○	●
	Lifting RPV into R/B with a large outdoor crane	○	○	○	○	○	○	○
	Deck plate construction	○	●	●	●	●	●	●
Increased efficiency of site work	All-weather construction	✖	○	●	✖	✖	●	✖
	Automatic welding	○	●	●	●	●	●	●
	Automatic cable pulling machine	○	●	●	●	●	●	●
	Automatic rebar assembly machine	✖	○	○	○	○	○	○
Reduction of site work load	In-factory assembly of reactor internals	○	●	●	○	○	●	○
	Prefabricated piping module	○	●	●	●	●	●	●
	Prefabricated rebar and steel structure module	✖	✖	✖	●	○	○	●
	Composite civil and mechanical modularization	✖	✖	✖	○	○	✖	○

○: Applied, ●: Expanded use of ideas, ✖ : Not Applied

construction method" applied to K-6, and the other is "Large block construction method" employed at K-7. The former originally came from Toshiba and Kajima, and the latter originated from Hitachi and Shimizu.

5.1 All-weather construction method

The efficiency of outdoor work at the Kashiwazaki-Kariwa site tends to deteriorate in winter due to adverse conditions such as heavy snowfalls and strong seasonal winds. The all-weather construction method has been designed to create a factory-like environment under such situation. In this method, steel structure for the R/B is assembled earlier than an ordinary schedule requires, then the building construction area is covered with a temporary roof and wall sheets to provide a favorable working condition. This method was first applied to K-2. Since then, it has been continuously improved through the successive construction of K-3 and K-6, for which main supplier was Toshiba.

For example, the covered area has been extended and the roof structure has been modified since its first application.

Figure 3 shows a schematic outline of the temporary structures. The temporary roof for the K-6 construction was able to be divided into four pieces for easy removal. Large components were lifted into the R/B by a tower crane with the roof opened. Construction materials such as rebars were carried inside efficiently with a diverse and multiple transport system consisting of monorail hoists, a temporary overhead crane, and wall cranes.

This method demonstrated its effectiveness at K-6 in securing efficiency, quality, and safety of site field work by shielding the area from wind, precipitation, and direct sunshine.

5.2 Large block construction method

Large block construction method has been designed to expand scopes of factory fabrication and concurrently executed site work. Components and structures are assembled or integrated into large block modules either at factories or at an assembly space on the site, and then they are lifted into their installation places so that construction work can be efficiently conducted. This method has enabled us to schedule several tasks proceeding in parallel.

A large crawler crane, with a lifting capacity of 930 ton at an operating radius of 30 m, was used for lifting the large block modules at K-7. The crane is capable of lifting the Reactor Pressure Vessel (RPV), the heaviest component of ABWR (840 ton with some internals when it was lifted), for direct carrying in and installation. Its maximum reach is 130 m (with 106 t) and it is capable of pro-

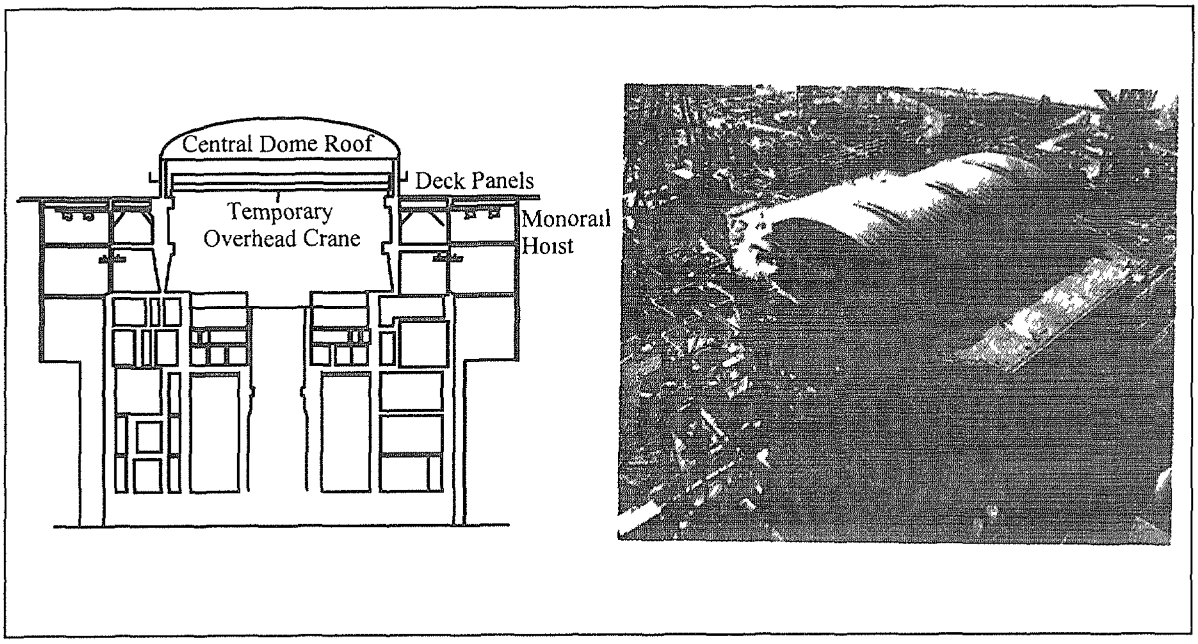

FIG 3 All-weather construction method at K-6

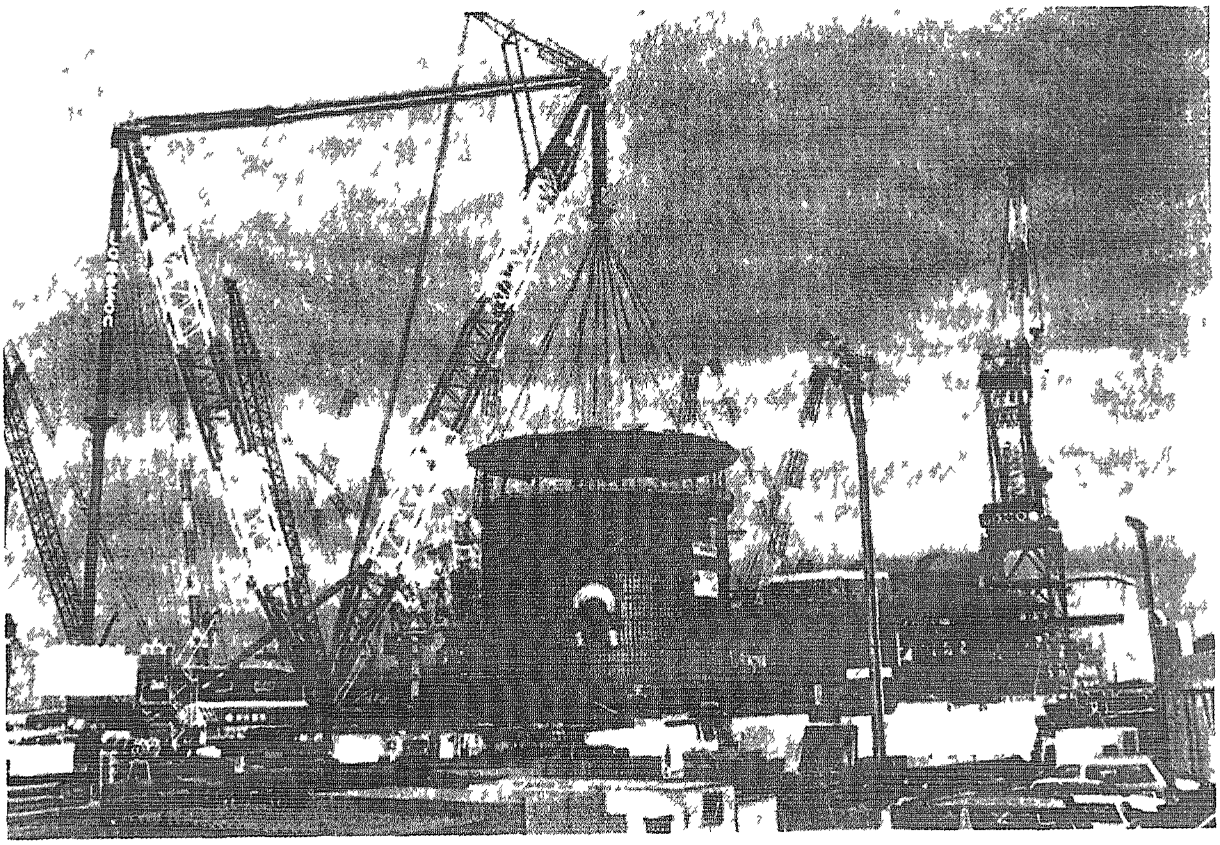

FIG 4 Installation of a RCCV liner module by a large crawler crane

pelling itself, and accordingly, one large crawler crane is adequate for the mechanical and electrical construction work of a plant if there are no restrictive conditions in the peripheral space of the construction area.

The following were major block modules at K-7:

(1) RCCV center mat rebar module (a 460-ton module).

(2) RCCV liner plate module (a 422-ton module as shown in Figure 4 and a 269-ton module).

(3) RPV pedestal module (a 560-ton module).

(4) RCCV upper drywell internals module (a 650-ton module).

(5) RCCV top slab composite module of rebar, liner and built-in piping (a 506-ton module).

The schedule for the RCCV erection constituted the critical path in the construction of K-7. And the modularization was applied to a maximum extent to the RCCV and its internals to shorten the critical schedule. The RCCV liner plates were modularized so that their installation would not cause any waiting periods in rebar work, and then the liner was used as a form for concrete placement. The RPV pedestal, the RCCV internals, and the RCCV top slab were assembled into modules to secure the concurrent progress of mechanical work and architectural work in parallel.

The effectiveness of this method was also demonstrated in smoothing out the site workloads. Because several assembly sequences were set in parallel using modularization techniques, the peak load of the site work could be controlled, which enabled us to utilize the work force effectively. Although the main objectives of the modular construction method lay in shortening the critical schedule, it was also applied to the sub-critical part of the construction scheme for K-7 because the application was evaluated to be beneficial from this point of view.

6. DETAILED ENGINEERING AT VERY EARLY STAGES OF THE PROJECT

There were two major reasons why early engineering was important in the K-6/7 project. One reason was to secure an appropriate period for step-by-step design reviews and verifications as explained earlier. The other was for the efficiency of the construction.

Well co-ordinated scheme of the building construction and the equipment installation is a key to any short construction schedules, therefore, the design for equipment, layout and building structure is required to be detailed at very early stages. This was especially important when applying the modularization to K-7, because components were brought into site earlier than conventional construction methods for pre-assembly, and because the modularization required more accuracy of interface conditions with other equipment and structures. Co-ordination of the interface conditions, which had been done by "on-the-spot" adjusting in old methods, was necessary to be resolved in a planning phase. Specifically, size of equipment, piping route, support design, and temporary installations were required to be determined very early.

Three-dimensional CAD systems were fully utilized for this purpose. The CAD systems were not only powerful nor useful in the design work, but also effective in the simulation of the construction work to prevent interference during installation of large modules.

7. CONSTRUCTION MANAGEMENT

TEPCO has continuously constructed nuclear power plants since its first in 1971. The lessons and learned in a construction project were reflected in its following project. In the construction of ABWR, TEPCO played the following roles utilizing the accumulated experiences.

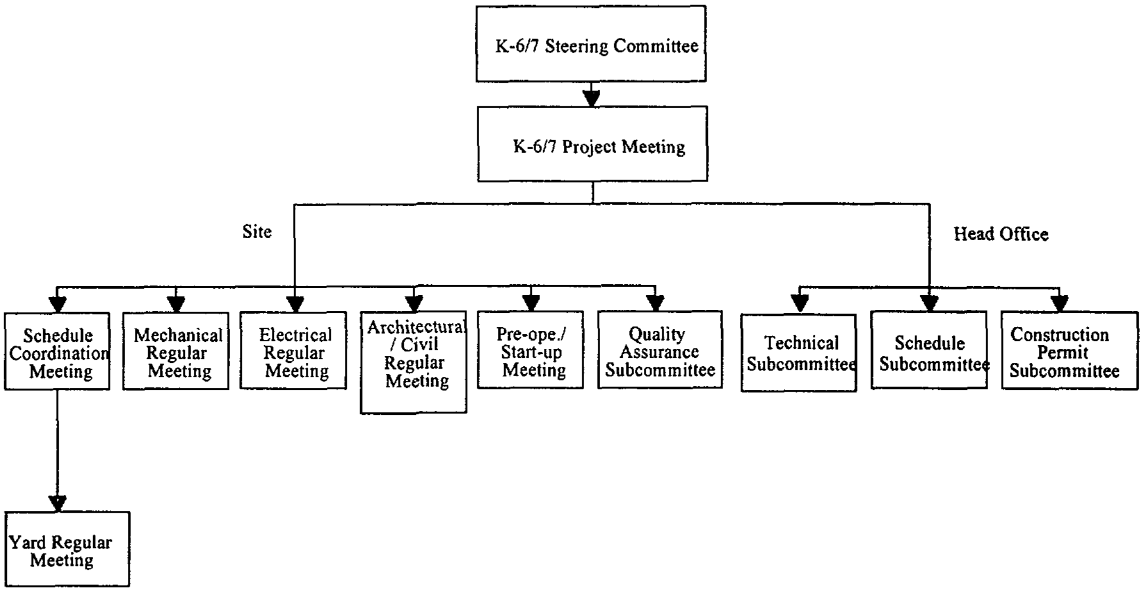

FIG. 5. Project organization for the K-6/7 project. [The K-6/7 Project steering organization consisted of members of TEPCO, TOSHIBA, HITACHI, GE, and consrruction companies.]

(1) Design related function including design change control, design review, and its verification.

(2) Project schedule control.

(3) Interface control between different suppliers.

(4) Quality assurance.

(5) On-the-job safety related function.

7.1 System and organization for the construction management

The contract for K-6/7 between TEPCO and a manufacturers joint venture of GE, Hitachi and Toshiba was, in principle, on a turnkey basis, and the joint venture supplied main systems except for those local systems whose interfaces with main systems were clearly defined. Such facilities were placed orders for with various suppliers on bid bases. While avoiding potential economical risks by the turnkey contract, TEPCO, however, was deeply involved in actual construction work because of the responsibility as an owner and operator. We believe that positive involvement of electric utilities is essential for good project control, better quality, well-organized activities of the suppliers, on-the-job safety, and so on.

A hierarchical committee organization was established as shown in Figure 5, in which TEPCO and the plant suppliers were participated to discuss engineering issues, to control the engineering and construction schedule, to co-ordinate various activities such as QA and licensing, and to review the progress of the project. Close communication between TEPCO and the plant suppliers was vital in the K-6/7 project, and it was held in these committees, joint task forces, and daily meetings.

7.2 Design related function

Design changes may have possibilities to cause unexpected results. We have made it a rule that new design is to be applied to a non-essential local area first, and then the application will be gradually expanded to other areas. For example, TEPCO had set a long-term strategy of adopting digital controls. It was first introduced in radioactive waste treatment systems, and then in the next project, BOP system became digital controlled. After reviewing the experience with these systems, TEPCO decided to introduce full digital control systems to K-6/7.

As explained earlier, the design review is a key to minimize the occurrence of troubles at new plants. In the K-6/7 project where many FOAK design features were adopted, TEPCO's initiative and suppliers' cooperation in the step-by-step verifications in each phase of the project progress was important to complete the construction as scheduled.

In addition, plant layouts were reviewed using CAD systems to confirm that the arrangements of equipment and piping satisfied the design criteria established by TEPCO to secure adequate maintainability and accessibility. The criteria stipulates such things as width and height of access routes, distance between adjacent pipes to secure appropriate working space for ISI, space around components, requirements for the operation stands, hooks over the components, and so on. The plant layouts and detailed structures of the buildings were also reviewed from the viewpoints of effective installation of equipment, piping, ducts, and so on. The CAD systems were utilized to simulate installation work of large block modules, and rework caused by the interference between piping and building structure was minimized.

7.3 Project schedule control

As explained in the previous section, the engineering for the construction of K-6/7 was required to start very early, and then, it progressed in sequence from the basic to the detail, from system design to equipment design, from plant layouts to construction planning, and so on. Step-by-step design verifications were conducted in a timely manner with the engineering progress. The schedule control over the engineering was one of the most important aspects of the construction management.

The plant construction schedules were also precisely controlled. A large volume of equipment and piping was needed to be brought into the installation places before installing the ceilings over the places, and materials were required to be delivered to the site on schedule. In addition to just reviewing the work progress, detailed scheduling in planning stages was more important than ever, because engineering, procurement, delivery, and construction were closely linked each other, and because the co-ordination of their schedule was essential in the new construction method.

In the planning phase, milestones of construction work were first set up, and then, a master schedule and a sub-master schedule were prepared based on the milestones. The sub-master schedule was a detailed construction schedule, which was prepared before the start of building erection, and with which actual work progress was checked weekly during the construction period.

7.4 Interface control

The interfaces between the systems supplied by different manufacturers were co-ordinated and reviewed jointly by TEPCO and relevant parties. In the K-6/7 project, TEPCO made a contract with the manufacturers joint venture of GE, Hitachi and Toshiba for the main facility. Although the responsibility to control the interfaces inside the joint venture lay on its participants in principle, TEPCO reviewed the suppliers' scope split, design interface, and schedule co-ordination results to minimize the risk of non-conformance and schedule delay. In addition, TEPCO conducted special audits at the site to check whether the interface control was appropriately done between suppliers.

7.5 Quality assurance

Suppliers' QA programs were reviewed by TEPCO before the start of manufacturing or installation work. Selections of sub-contractors by main suppliers were required to be accepted by TEPCO. If a sub-contractor did not have any past experience of nuclear business with TEPCO, its technical capability and quality level were evaluated based on the documents submitted by main suppliers.

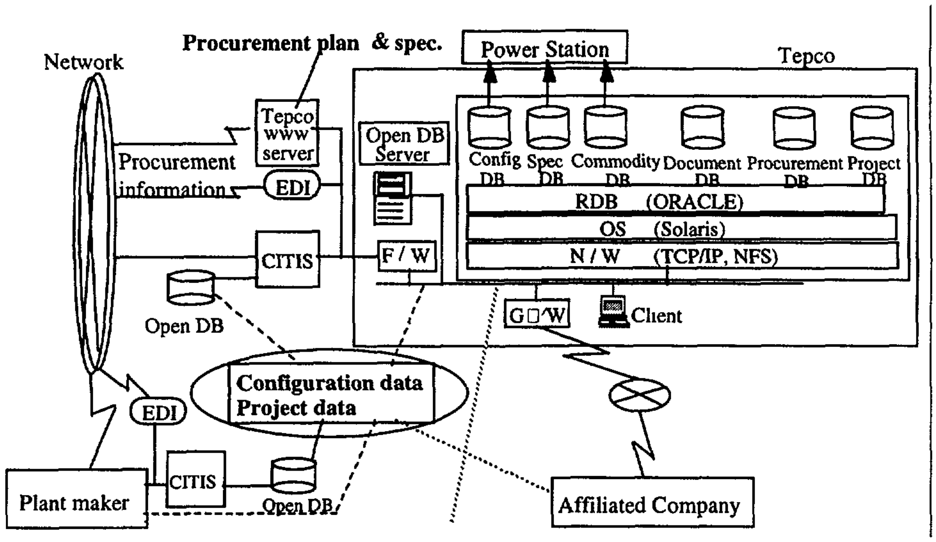

FIG. 6. An image of information systems for the next construction project.

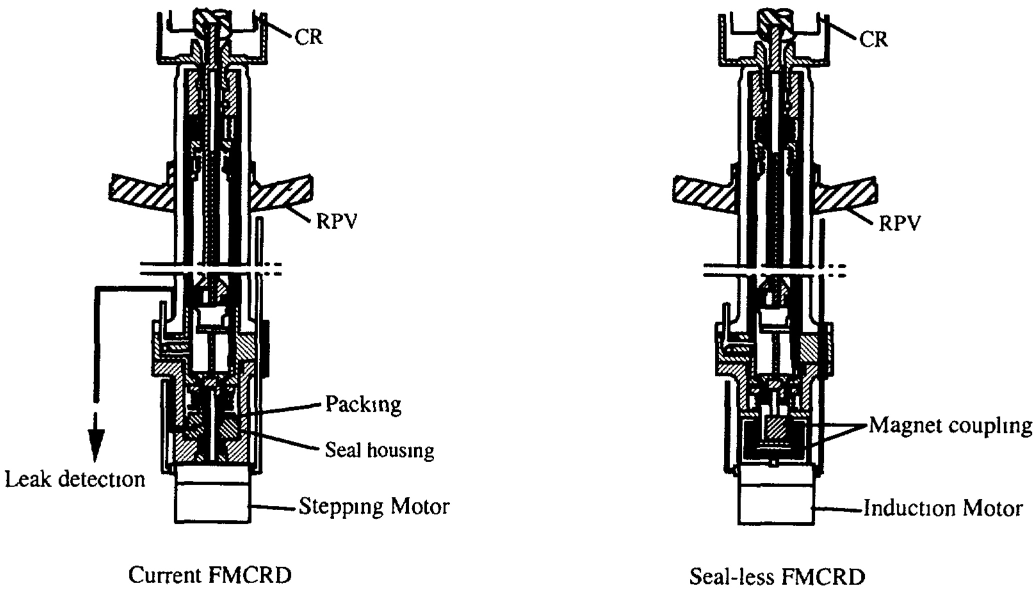

FIG. 7. An example of further improvements: Seal-less FMCRD.

The status of quality assurance activities at factories and the site was periodically checked through QA audits and QA patrols. QA audits were conducted on all the main suppliers and many new sub-contractors including international companies.

TEPCO's witness inspections and governmental pre-use inspections were conducted at factories and the site. For the main steam turbine generator and some reactor internal components that were manufactured by GE for the first time in nearly twenty years for TEPCO, it dispatched its engineers who stationed at GE's factories conducting witness inspections, daily monitoring and in-process quality surveillance.

In addition to the inspections on system and component bases, overall plant check-ups were conducted at three stages of construction; before the hydrostatic test of Reactor Pressure Vessel at the site, before fuel loading, and before the start of commercial operation. The purpose of the check-up was to ensure that the facility was properly installed from the operation and maintenance standpoints and no major issues were left out.

7.6 On-the-job safety

TEPCO evaluated on-the-job safety prior to the execution of such hazardous work as handling of heavy components. A total of 11 work plans for K-6 and a total of 7 work plans for K-7 were subjected to the safety evaluation. Those included were, for example, the work plan for the installation of the RCCV liner modules and the work plan for the installation of the lower bodies of the main condenser. We also did our utmost to assure the safety through daily safety patrols.

8. FURTHER IMPROVEMENT TO NEXT ABWR CONSTRUCTION

TEPCO is now making every effort for further improvement to next ABWR construction projects. The followings are our focal points of the improvement.

8.1 Standardization

ABWR has been developed as an international standard BWR. TEPCO and almost all of the other BWR owners in Japan have plans to install ABWRs. Taking advantage of this situation, design change from K-6/7 should be minimized in order to reduce engineering cost. When we use the same design, that means we can define every detail of the plant at the start of a project. This is very beneficial because it enables us to seek more possibilities for cost reduction such as further improvement in the construction methods or the procurement of separate equipment and systems from more competitive market based on well-defined interface design information.

Construction management is sought to be standardized. Every electric power company tends to have different ways of the management, and sometimes this causes additional work at suppliers. TEPCO has been encouraging other utility companies to standardize it, and such things as document control routine have already been established as standards for next projects. Construction management will be further enhanced and streamlined by applying the latest information technology to standardized management work. It will enable us to make the best use of the data and the knowledge accumulated through the construction of the first ABWR. Figure 6 shows an image of the information system that is now being studied for next construction project.

8.2 Rationalization and simplification of the plant facility

Although the design change from K-6/7 is to be minimized in next projects, a few modifications are now being discussed. Candidate ideas of the design change must meet the following conditions:

(1) Benefits gained can offset the engineering cost penalty.

(2) New design is able to become a standard.

For example, gland packing of FMCRD is planned to be replaced by a seal-less magnet coupling as shown in Figure 7, and its stepping motor will be replaced by an induction motor. They have enough benefits to offset the cost penalty because the seal-less coupling eliminates the leak detection piping and the surveillance camera, and because the induction motor is cheaper than the stepping motor. The design is now being discussed and tested as a candidate standard in a joint study program of the Japanese electric utilities and manufacturers.

8.3 Further improvement to construction method

The construction schedule is sought to be reduced by improving the construction methods. Based on the data accumulated in the K-6/7 project, we believe that the construction duration will be less than 45 months without major changes of the construction methods. And it will be less than 43 months when we improve the large block modular construction method further and enlarge the area of its application in next projects.

9. CONCLUSION

The first ABWR units, Kashiwazaki-Kariwa #6 and #7, have come into the world as scheduled, fulfilling every development target without major troubles. The construction duration was shorter than that of a conventional BWR in Japan by nearly one year. TEPCO believes that design features of ABWR for better constructability, prudent design change control with the principle of "test before use" applied to the FOAK design, advanced construction technology, detailed engineering at very early stages of the project, and good construction management were keys to this success. The positive experience in the construction and project management of K-6/7 will be standardized for further improvement to next ABWR construction projects.

KEYS TO ECONOMIC VIABILITY OF EVOLUTIONARY WATER COOLED REACTORS

(Session V b)

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IMPLEMENTATION OF UTILITIES OPERATION AND MAINTENANCE EXPERIENCE INTO THE EUROPEAN PRESSURIZED WATER REACTOR DESIGN

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Abstract

Since 1992 Electricité de France EDF and German Utilities GU work together with Nuclear Power International NPI, a subsidiary of Framatome and Siemens, in the development of the future European Pressurized Water Reactor EPR. The EPR is an evolutionary concept, based on the French N4 plants and the German KONVOI plants. From the beginning, experienced operation and maintenance people from the precursor plants participate at the design process. Their experience will lead to a plant, which is not only characterised by low investment costs, but also by good operability, high availability and low operation and maintenance costs. No expensive back-fittings should be necessary after commissioning, to reach these availability and maintenance targets. The utility specialists give design requirements for outage performance, system design, and layout. These design requirements are really determining the system performances, and not what was design basis before. It does not necessarily lead to system increases. Mainly it is a shifting of the emphasis to other items. There are even cases, where the system performances can be reduced. Mostly very small modifications, which are nearly cost neutral when implemented early in the design, have big impact on the further operation. If there are big cost influences, a sound balance between investment and gained availability is made together with the designers. There is very fruitful discussion between designers and operators, which is highly estimated by both sides. In this frame also new, revolutionary ideas are coming up, which are going mostly in the direction of investment cost reduction, without loosing operation freedom. It is the first time in Europe, that designers and operators are working so close together. It is also the first time, that the management and the decision making is dominated by the utilities.

1. INTRODUCTION

Since 1992, Electricité de France EDF and German Utilities GU work together with Nuclear Power International NPI, a subsidiary of Framatome and Siemens, in the development of the future European Pressurized Water Reactor EPR. The EPR is an evolutionary concept, which is based on good practices and designs in France and in Germany, combined with enhanced safety features. Also new ideas are implemented, where weak points are identified. The recent precursors are the French N4-plants and the German KONVOI plants. The economical success strongly depends on high availability, combined with low operational and maintenance costs. These targets must be in a sound balance with the investment.

After commissioning of the current plants, the utilities made big efforts in terms of modification of system design and implementation of new practices, to enhance availability and reduce operation costs. The vendor companies were de-coupled from this post-erection process and have only little experience in the field of operation and maintenance. Indeed, this double process, first construction, then stepwise improvement during commercial operation is not the most effective way to save costs. All retrofitting measures including loss of availability had to be paid by the utilities and it lasted several years to reach good results in time availability.

For the EPR, the intention was to implement the operator experience from the very first beginning of design, to avoid any wrong investment and to implement beneficial features. From EDF side, a special working group - CIDEM (French acronym for Design Integrating Availability, operating Experience and Maintenance) - was founded. All members of this group have practical maintenance and operation experience and participate at the EPR working groups. From German Utility side, operation and maintenance people, outage planners from operating plants work close together with CIDEM and participate in the most important EPR working groups, like system design and layout. These French and German operation engineers issued requirements for outage and operation, which are the basis for the design of the EPR. It came out, that French and German Utilities have nearly the same thinking. Their harmonisation process was easy. Due to their common position, they had, and still have, a big influence on the technical options taken in the EPR project.

2. FIELDS OF IMPLEMENTATION OF OPERATOR EXPERIENCE

2.1. Reduction of unplanned unavailability

2.2. Reduction of planned unavailability - outage reduction

2.3. Reduction of investment costs by proper system design and layout

2.4. Optimisation of the operation and reduction of operation costs

2.5. Optimisation of the maintenance and reduction of maintenance costs

2.6. Dose reduction, implementation of the ALARA principle

In the following chapters, examples for each field are given. It is impossible, to describe the full scope in a short way.

2.1 Reduction of unplanned unavailability

2.1.1 General

Unplanned unavailability means reactor trips or power reductions and the duration when less or no energy is produced.

2.1.2 Limitation system

The comparison of reactor trip occurrences and their induced plant unavailability in France and Germany showed, that there was a clearly identified advantage to introduce the German concept of limitation in a new plant design. This limitation system is acting with staggered measures in addition to the controls in order to avoid when possible to reach the protection set points and to avoid reactor trip actuation. The today existing German limitation system has been analysed in detail by the French and German engineers and an optimised set of functions has been retained based mainly on the feedback of experience.

2.1.3 Degree of automation

The degree of automation has been fixed by comparison of the French and German existing solutions. All elements, which have influence on availability will be automated in the future EPR (e.g.: automatic start of the third main feedwater pump when required, automatic start of main condensate pumps).

2.1.4 Fixed in-core instrumentation

KONVOI plants are equipped with a permanent in-core instrumentation, the N4 plants have a very accurate movable in-core instrumentation with long time response and rely on a less accurate ex-core instrumentation. With the fixed in-core measurements, periodically calibrated by the movable aeroball system, the on-line core surveillance can be more precise. Therefore, the operation restrictions can be reduced (e.g.: PCI constraints) and the required operation margins to protection actions can be optimised. Operation gets easier and the risk to act a core related protection is reduced.

2.1.5 Digital I&C - Computerised control room

The French N4 plants are equipped with digital I&C and a computerised main control room. The latest German plants are still based on hardwired technologies. The new French concept gives advantages in terms of plant surveillance, better overview of the operators, faster capability to restart, more flexible use of computerised procedures, more freedom for automation, more flexible man machine interface. The good French experience is implemented in the EPR.

2.1.6 System design - Nuclear island

The degree of redundancy of operational systems, which are important for availability, was questioned. A typical example is the number of coolant evaporators. French plants have two units, in Germany there is only one. The good field experience in GU plants showed, that only one unit is sufficient, but the active components like pumps and control valves should be doubled.

2.1.7 System design - Conventional island

The Conventional Island is, up to now, separately developed in France and in Germany. Nevertheless in each country the operation teams have also strong influence on the design. Major point is here also the degree of redundancy to be provided for systems and components which are important for availability. A good example is: how many main feedwater pumps to implement? On coal-fired plants, the trend is to have 2 x 60 % pumps with capability for overload, that means one pump for a certain time up to 80 % with speed control. Experience in nuclear power plants shows, that there were several cases, where the 3 x 50 % design avoided power reduction and partial trip. Considering the big sizes to implement for a 1750 MWe unit, the today solution for EPR is 4 x 33 %. So far, pumps which are in operation today, can be used.

2.2 Reduction of planned unavailability - outage reduction

2.2.1 General

The outage duration is the main contributor to the overall plant unavailability. Former general maintenance outages had a duration of about 30 days. In the last years, the KONVOI plant succeeded in outage reduction with maintenance down to 15 days. This reduction has been performed with modification of systems, tests, practices. There was no reduction in plant safety.

Several French CIDEM specialists participated in these outages. Together with the German outage planners, a general time schedule for EPR was developed, which is the basis for the EPR system design. In addition, a set of operator outage requirements has been fixed. The outcome was that these requirements are really driving the system performances which was not the case for the past designs. In some cases, the system performances could be even reduced.

2.2.2 Outage phases

The outage can be roughly separated into three phases:

- The shutdown sequence, which comprises the following sub-phases
  - power reduction
  - boration
  - cooldown
  - mid-loop operation with nitrogen flushing
  - opening of the vessel and flooding of the reactor cavity

- The work phase, which comprises the following sub-phases
  - removal of the vessel internals
  - core unloading
  - core loading

- putting in the vessel internals
- works on safety trains

- The start-up sequence, which comprises the following sub-phase
  - lowering the water level in the cavity/vessel to mid-loop
  - closing of the vessel
  - vacuum venting
  - filling of the reactor coolant system
  - degasification
  - pressurisation
  - start of reactor coolant pumps
  - heat up of the reactor coolant system
  - deboration and power increase

Each phase and sub-phase includes a set of periodic tests, which can be only performed during this plant state. Based on experience, for each sub-phase a maximum duration was fixed. This maximum duration determines the system performance, e.g. the capacity of the residual heat removal system for cooldown. When consideration of a shorter outage duration would have induced higher investment costs, a sound balance has been found. But most of the benefits in time have been reached by implementation of small details, which are nearly cost neutral because they are taken into account at an early design state.

In the following, there are some examples of how to improve the outage performance.

2.2.3 Degree of automation

In former plants, the cooldown is performed manually. For being not too fast and violating technical specifications, the operators have to operate with a temperature gradient, which is much lower than what is technically allowed. In EPR, this cooldown will be automated. During cooldown and start-up, auxiliary systems, like coolant evaporators, degasifiers, have the be started and stopped depending on the water management status. These sequences need a lot of operator time and attention, while he should focus on items in the overall plant shutdown or start-up process. Therefore, such kind of sequences will be automated in the EPR. A lot of time critical tests, like tests on the pressurizer and steam generator safety valves have to be performed. These typical test sequences are planned to be done in automatic. In general, functions are automated, when

- they are time critical during shutdown and start-up,
- there is a investment risk, e.g. start/stop of big pumps,
- they disturb the operators during other important work or attention,
- it is a boring and repetitive work, which is a potential for human failure,
- it is important for safety,
- it is important for availability.

These are quite different criteria for automation, when compared with those used by the designers before.

2.2.4 Full-flow degasifier in the chemical and volume control system

During cooldown, hydrogen and noble gases have to be removed from the primary circuit before opening. This degasification was formerly done by spraying in the volume control tank and was always significantly time critical. KONVOI plants are equipped with a full-flow vacuum degasifier which drastically reduces the required time. It is also used to remove oxygen during start-up and avoids the injection of hydrazine. The same concept has been implemented in the EPR.

2.2.5 Mid-loop operation

Mid-loop operation is a plant state with reduced water inventory and core loaded. Reduced inventory is necessary for opening of the vessel head. Mid-loop is advantageous for flushing the reactor coolant system gas phase with nitrogen and air before opening, to reduce the radiological impact on the workers inside containment. Operators opinion is, that mid-loop operation is necessary and can be a good practice but the safety in such a situation, with reduced coolant inventory, must be guaranteed. The retained solutions to reach this required safety level are accurate, reliable and redundant mid-loop level measurements, automatic mid-loop level control and automatic makeup with the safety injection system in case of level drop.

2.2.6 Nitrogen flushing during mid-loop operation

Before opening of the reactor coolant system, the gaseous phase is swept first to the gaseous waste processing system and afterwards, if the noble gas concentration is low enough, to nuclear ventilation system. Such kind of operation has been foreseen in German plants, since there is a vacuum venting system. But the procedure was not very effective. The operation engineers found for the EPR project new injection and suction points and increased the flow rates. With this, the time critical duration has been reduced. The remaining amount of noble gases in the reactor coolant system is minimised. Releases in the reactor building during opening of the reactor coolant system are mitigated.

2.2.7 System capacities for draining of the reactor cavity

After core unloading and after core reloading, about 1800 m^3 of borated water has to be drained from the reactor cavity to the refuelling water storage tank. In order to avoid contaminating the tank, this flow has to be routed via resin filters. The capacity of the normally used fuel pool purification filter is about 30 kg/s.

This would lead to a time critical drainage time of 16 hours, twice per outage. Parallel use of the coolant purification resin filters doubles the flow rate and halves the required time. The connection lines have been implemented in the EPR design.

2.2.8 Vacuum venting before filling of the reactor coolant system

After tightening of the reactor coolant system during mid-loop, there is still a free air volume of about 200 m^3 inside. During filling and pressurisation, this air volume is compressed and is concentrated at the highest points. It has to be vented during pressurisation and after several short starts of the main coolant pumps. The main concentration points are the vessel head and the U-tubes of the steam generators. A big part of the air gets dissolved in the coolant. The oxygen content increases. This is the situation in former German plants and in France. Two problems are coming up. The venting process is time consuming and during heatup, oxygen limits in the coolant have to be respected. If a lot of air is dissolved during pressurisation, the degasification process lasts longer. Hydrazine has to be injected for oxygen elimination. The EPR will be equipped with a vacuum pump, which evacuates nearly all the air during mid-loop operation, before filling of the reactor coolant system. This system is intended to be kept in operation also during filling. The remaining air content will be small and the venting procedure can be drastically reduced. Only a very little air amount gets dissolved and the final oxygen content will be very low. Even without hydrazine injection, there will be no more start-up delays to respect the temperature dependent oxygen limits.

2.2.9 N+2 Safety system design

In current plants, the French safety systems are designed 2 x 100 % (N+1), the German safety systems are 4 x 50 % (N+2). From the first point of view, the 2 x 100 % concept requires less components and looks therefore cheaper. For a 4 x 50 % concept, even two additional building compartments have to be implemented to separate the trains. After more detailed analyses of the induced investment, it has been found out, that the costs are not much higher with a 4 train concept. The components are smaller, more standardised equipment can be chosen and some important big pipes like headers can be deleted. On the other side the safety is somewhat improved considering failure risk. The overall capacity of 2 x 100 % and 4 x 50 % is the same.

Considering the subject of this paper the 4 x 50 % concept, commonly described as the N+2 concept, induces in addition the following advantages:

- **On power maintenance.**

  The N+2 concept allows to consider the maintenance of one train during power operation and to postulate in addition a single failure in case of an accident without impairing the safety criteria of the safety analysis.

  On power maintenance can therefore be performed with the following advantages:

  - On power maintenance can be carefully planned and carried out with the best staff in a free time window over the year.
  - On power maintenance reduces the workload during outage and makes the outage planning more save.
  - On power maintenance reduces the maintenance costs, because the work is decompressed and more can be performed by own staff.

- **Increase of admissible repair times.**

  The degree of redundancy is higher. Therefore the probabilistic risk assessments allows longer repair times during power operation. The repair can be performed with a better preparation and perhaps with more qualified staff.

- **Works on safety trains during outage.**

  Also during outage the N+2 concept allows a more flexible planing of maintenance work on a safety train. More trains are kept available. Due to the time dependant decrease of residual decay heat, the required capacities for heat removal are decreasing too. After a certain time, the 4 x 50 % configuration becomes automatically a 4 x 100 % configuration considering heat removal. Maintenance is possible in phases, where it is forbidden in case of 2 x 100 %. In certain plant conditions, maintenance can even be performed on two trains in parallel. The combination of these two trains is nearly free. A degree of redundancy for residual heat removal of 2 x 100 % or 3 x 50 % can be kept all the time. Even with more trains, the required time for safety train maintenance can be reduced.

- **Cost savings considering spare parts.**

  With 4 x 50 %, the single components are smaller and less costly. Therefore, the costs for the spare parts in the store are also lower.

2.2.10 Maintenance provisions on electrical busbars

The electrical house load grid is separated in 4 trains. Each train comprises of an operational part and a safety part. The safety part supplies the safety trains and is fed by the operational part or by the diesel generators. The operational part and the safety part of one train must be maintained together. During this phase the assigned mechanical safety train is not available. But there are other consumers, which have to be kept in operation, like safety lighting, fire detection and fire fighting, site security, communication equipment, lifts, hoists, coolant evaporators, waste water evaporators. In former designs, no provisions have been made for these consumers. The operators had to keep them in operation with a lot of mobile cables through the corridors. Even if these consumers were doubled, their alignment to the busbars was so unfortunate, that with maintenance on two trains, the whole function was lost. Another restriction is, that the provisional connection must be from the same quality, related to diesel supply. A lot of back-fittings have been performed in former plants to relax this situation.

For EPR, the operators identified the sensible equipment. They fixed the allowed maintenance combinations of electrical busbars and put the consumers on the right busbar combinations, to keep the function under all conditions during outage. For some equipment, double electrical supply has been foreseen from busbars, which are never isolated at the same time. Special busbars have been implemented, which are kept under voltage during the whole outage. Their maintenance is performed during power operation.

2.2.11 Accessibility of the reactor building before and after outage

In German plants the reactor building is permanently accessible during power operation. This is mainly due the inside spent fuel pool. In French plants, where the spent fuel pool is located in a dedicated building, the reactor building is not accessible during power operation. All ventilation penetrations are closed. This is a basic safety feature, which is preferred by the authorities. Some time critical outage works to be done in the reactor building should better be performed before or after outage, e.g. tests of the polar crane, tests of hoists, which are permanently used during outage. Considering these constrains, the less time critical these operations are, the shorter the outages will be. The French and German operation engineers defined the amount of work to perform before and after outage and identified the time needed for this work considering a reasonable limit of the number of workers allowed to enter the containment. With this the accessibility could be limited to 10 days before and after outage. The work location were clearly defined and the required shielding will have to be adapted accordingly.

2.2.12 Lay-down areas in the reactor building

Experience shows, that in outage, there is always a lack in storage space for heavy tools and vessel head equipment inside the reactor building. Each part is used in a dedicated sequence and must be accessible from the polar crane. A very detailed planning for these movements has to be performed, to avoid additional crane transports or not to block the sequence completely with direct influence on the outage length. The tools used in France and in Germany are different. The first task in the project was to get information about the partners' tools and to finally harmonise a set of tools for the EPR. Afterwards, a handling sequence for the vessel head equipment and for the reactor internals has been defined. This was the basis for the designers to create the EPR laydown areas inside the reactor building.

2.2.13 Hatch opening and closing

French plants are equipped with an equipment hatch, German plants with an equipment airlock. Design basis for the EPR is an equipment hatch. The EPR will be a standardised reactor, with several units on one site. One multi-stud tensioning machine will be used for several units. Therefore transport inside and outside the reactor building is essential. These dimensions are only possible with a hatch. The containment is open, when the hatch is open. The multi-stud tensioning machine has to be transported in before opening of the reactor coolant system. The hatch must be closed, when the reactor coolant system is open and when there is reduced water inventory. Several other transports have to be performed, which are critical when not possible at the right time. Derived from all possible outage sequences, the required opening times and duration's and the allowed time for opening and closing of the hatch have been defined and checked considering the confinement requirements.

2.3 Reduction of investment costs by proper system design and layout

2.3.1 General

A large field for cost savings is concerned by the design of the nuclear auxiliary systems and not the safety systems. They are quite different in France and in Germany. Each of the two designers knew only their own design. They were not able to harmonise themselves. It was a utility operator task, to make the best choice.

2.3.2 Optimisation of the coolant storage capacity

In France there are dedicated tanks for letdown coolant from the primary circuit and for demineralized water after evaporation. With this, the slightest contamination of the demineralized water with boron can be avoided. The deboration of the reactor coolant system can become more effective. But this leads to a high tank storage capacity. The German plants use 6 coolant storage tanks, interchangeable for letdown coolant and demineralized water but these tanks cannot be drained completely and potential demineralized water pollution can occur. After long discussions and calculations between utilities, it has been found out, that this potential pollution was in any case very limited and couldn't have big impact on the deboration of the primary circuit. The combined tank solution has been finally chosen for the EPR .

2.3.3 Gaseous waste processing system

The treatment of waste gas is completely different in France and in Germany. In France, all tanks with a possible contaminated gas surface are connected to an evacuation system which pumps the gases into big decay tanks. After decay of several weeks, the content is measured and manually released to the stack. In Germany, the waste gas system is a closed circuit. All tanks and evaporators are connected to this system and are permanently flushed with nitrogen at a slight under-pressure to take into account the protection of the operating staff. A part of the flushing gas is routed to charcoal beds, where the noble gases xenon and krypton are delayed much longer then the nitrogen. Due to the charcoal, the gas storage capacity is increased. If the pressure in these delay beds increases, there is an automatic release of the delayed gas to the stack. The advantage of this last conception is, that it induces less gas releases. All volumes are connected to the same system. If one tank is drained, another one is filled. The overall gas volume is kept constant. The required storage volumes are smaller, fewer building volume is required. A drawback, compared to the French solution, is the automatic release, which is currently not licensed in France.

The final choice was made for a reduced circuit system, where all tanks are flushed and a fewer number of delay beds are used as breathing storage tanks, with floating pressure. With this, also the French practice with manual release can be realised.

2.3.4 Fuel pool outside the reactor building

In France, all fuel pools are outside of the reactor building. The reactor building is not accessible during power operation. In Germany the fuel pools are inside the reactor building. The outside pool has the advantage, that the size of the pool can be changed and the shielding inside the reactor building can be reduced. Even the overall size of the reactor building can be reduced. It is now driven by the accident needs. An outside pool has been chosen for EPR.

The outside fuel pool allows the docking principle for the spent fuel cask. The cask is watertight coupled below a dedicated cask loading pit, which has a direct connection to the fuel pool. This French solution is preferred by the operators, because the cask is kept dry and is not contaminated. It must not be washed afterwards. The cask handling sequence in simpler. In current German plants the cask is lowered directly in a water filled pool beside the fuel pool. Contamination protection is performed by a dedicated shirt, which is filled with demineralized water.

2.3.5 Separate independent waste building

In German plants, the waste storage and treatment - liquid and solid - is located in the nuclear auxiliary building. This means that each unit has the whole equipment. In France, always several units are located on one site. They have a common building for waste treatment. This waste building has its own electrical supply and separate cooling means. It is not impacted by outage tagging. This is a big advantage, even if only a single unit is built. Therefore, for EPR it was an easy choice for the operators to follow the French design. The only important item is, that there must be a subsoil corridor connection between plant unit and waste building. It must be possible, that the people can change the buildings and make transports without leaving the controlled zone.

2.4 Optimisation of the operation and reduction of operation costs

2.4.1 General

A lot of items, which are indicated in the previous chapters, reduce already the operation costs. Pure operation costs are costs for fuel, consumption material and costs for the staff.

2.4.2 Fuel costs

The core for the EPR is designed based on plant experience with extreme low leakage and high enrichment. A further step can be the optimisation of the fuel cycle and the burnup by short cycles. This know-how is purely on the utility side.

2.4.3 Consumption material

The major cost impact has the consumption of boron which cannot be recycled. There are big differences between France and in Germany. The cost impact is even higher, when enriched boron is foreseen, like in the EPR. Weak points have been found in France and in Germany. The vent and drain system has been optimised, to make it possible to recycle all boron.

2.4.4 Man Machine Interface

Target is, to operate the plant with a reasonable number of staff. A potential reduction of personnel shall not lead to drawbacks in safety and availability.

The number of operators in the main control room can be reduced by an optimised man machine interface. The EPR shall be operable with 3 persons in the main control room. The digital control room opens a large field of new possibilities. EDF has long lasting experience on simulators and on the recent N4 plants. This experience was gained with operator participation from the very first beginning. The development process is going on in the frame of the EPR with strong operator contribution. An improved alarm concept is aimed to a better overview of the plant conditions. The degree of automation is strongly linked to the required number of personnel and to plant availability.

2.5 Optimisation of the maintenance and reduction of maintenance costs

2.5.1 General

Maintenance is linked to the work on the components themselves, the space around the components, their location and location and space for workshops. Another item is the possibility to make adequate system tagging, without significant influence on the overall plant. All this is a typical field, where the utilities are competent. The input is coming from the operators.

2.5.2 Choice of the components

Proven and standardised equipment is chosen. The number of different components is minimised, to limit the spare part stock. Industrial standards are used as often as possible.

2.5.3 Arrangement of the components and space around them

The components are arranged in dedicated position and in a dedicated height above the floor. Depending of the weight of the pieces there are hoists. There must be sufficient space for maintenance around and there is dose and temperature protection for the workers. All these design conditions are fixed in a set of 'Layout requirements' which are created together by designers and operators.

2.5.4 Work areas and workshops

The needed space, the preferred location and the needed equipment in terms of crane capacity of each kind of workshop has been given by the utilities, based on French and German experience. The harmonisation process between French and German operators was difficult, because the maintenance strategy is different. EDF relies more on centralised national workshops, while in Germany, with several companies, more work is performed on site. The designers tried to implement these surfaces in the layout without increase of the buildings.

A new, advantageous feature is the space in front of the hatch on top of the fuel building, and just beside, the large free area on top of the nuclear auxiliary building. These surfaces in the range of 1000 m^2 belong to the controlled zone, are equipped with cranes and can be used for storage of heavy equipment, as well as for workshops. Due to the different maintenance policy, the surface distribution to different tasks could not be fully harmonised between French and German Utilities. For German Utilities, some space must be reserved for decontamination and maintenance of reactor coolant pumps, while this work is performed in France in a national workshop. Nevertheless the pure overall space requirements for the future EPR are the same for EDF and German Utilities.

2.5.5 Possibility for system tagging

For maintenance or repair, whole systems or parts of systems have to be isolated and drained. To make these tagging easier and to make isolations during power operation possible without loosing the whole function, the systems must adapted for tagging. Isolations and re-qualifications in adequate zones must be possible. Each system drawing is checked with special emphasis for tagging possibilities by experienced outage and maintenance preparation people. Special valves to ease isolations are added.

2.6 Dose reduction, implementation of the ALARA principle

2.6.1 General

Dose reduction means low activity inventory in the systems, proper possibilities for system cleaning before maintenance, proper system arrangement in the layout, adequate shielding, and reduction of the amount of maintenance work. The ALARA principle - as low as reasonable achievable - is a sound balance of all these aspects.

2.6.2 Low activity inventory in the systems

The source term depends on the choice of materials in the primary circuit, mainly in the steam generators. Over the years, lot of experience has been collected on this field. Important is also to use as far as possible alloys with the lowest potential to generate activated Cobalt.

2.6.3 System flushing possibilities

After isolation of system parts for maintenance, there must be possibilities to flush them with demineralized water or even with chemical reagents. The system must be designed for this.

2.6.4 Proper system arrangement in the layout, adequate shielding

The components must be arranged in a way, that irradiation of maintenance people from other sources around is avoided. Non radioactive components should not be located in rooms with radiation fields or spots. The back of the working people must be kept free of irradiation sources. Adequate shielding has to be foreseen in maintenance areas. Fixed shielding is preferred compared to mobile shielding. The proper appliance of these basic principles is surveyed by the French and German Utilities within the Layout Working Group.

2.6.5 Reduction of maintenance work

One of the best ways to reduce the dose is not to do the work. Based on former experience, the maintenance intervals should be increased as far as possible. The fact should also be considered, that a lot of damage can be caused by the maintenance itself. A good way in this direction is the reliability-centred maintenance, which is developed by EDF. Developing of a new maintenance strategy is undertaken under pure utility responsibility.

3. SUMMARY

From the beginning of the conceptual phase, the operators have big influence on all fields, where they are competent. Such fields are operation, system design, layout, outage, maintenance. Their experiences are highly estimated in the project. All these items are fields, which are very important for the future economical success of the project. This leads to an operable plant and to high availability from the very beginning of operation because the operation requisites have been considered at the very beginning of the EPR design. It is the first time in Europe, that designers and operators are working so close together. It is also the first time, in the management and the decision making, that the utilities have the possibility to directly act on the design choices.

FEEDBACK OF OPERATION AND MAINTENANCE EXPERIENCE INTO EVOLUTIONARY LWR DESIGNS IN THE USA

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Abstract

U.S. utilities, with extensive support and participation from several international companies, have led an industry-wide effort in close cooperation of the U.S. Department of Energy (DOE), to establish a technical foundation for designing the next generation of light water reactors, referred to as Advanced Light Water Reactors (ALWRs). The cornerstone of this effort is the ALWR Utility Requirements Document (URD). ALWR requirements are driven by utilities, but with broad industry participation, including U.S. nuclear steam supply vendors, as well as engineering service, consulting, architect-engineer, and construction companies. Thus, there is essentially a consensus of the industry as to those features to be sought in the next generation of plants, based on the information and lessons learned from over 35 years of operating over 100 LWRs in the U. S., and many more internationally. The URD addresses the entire plant, including nuclear steam supply system and balance of plant. The requirements are intended to provide improved, standardized versions of ALWRs, which eliminate most of the issues and inefficiencies associated with some of the existing designs; assure a simpler, more forgiving plant design that is excellent in all respects, including safety, performance, constructibility, and economics. The U. S. Nuclear Regulatory Commission (NRC) is directly involved with the URD and has published a Safety Evaluation Report on the requirements for each type of ALWR. Through the NRC review, the URD supports improved stability in the regulatory basis for ALWRs by including agreements on the outstanding licensing and severe accident issues. Looking forward, the URD provides a set of utility technical requirements, which can be used in an ALWR investor bid package for the detailed design, licensing and construction. The resulting bid package will provide a basis for investor confidence in implementing an ALWR.

1. INTRODUCTION AND BACKGROUND

EPRI began the activity to develop a new generation of LWRs in 1983 through the establishment of preferences and prerequisites of utility executives for ordering new nuclear power plants. Utility executives indicated that new plants must be safer and simpler and must have greater design margins. They also supported making improvements to established LWR technology, rather than developing new reactor concepts requiring prototype demonstrations. Furthermore, they wanted the option to build mid-size nuclear plants in addition to large sized plants. It was judged that midsize plants could better support the demands of slower load growth, and would more readily accommodate the introduction of safety features that rely on natural forces, thereby providing additional opportunity to simplify plant design and operation.

In response to these industry guidelines, EPRI initiated the ALWR design effort in 1985 by beginning development of the ALWR Utility Requirements Document (URD)[1]. In addition to providing a comprehensive set of utility design requirements, the document was to address over 700 regulatory issues required to be resolved for future designs by the U.S. Nuclear Regulatory Commission (NRC). The first three volumes in the URD were completed in 1990 and contain more than 14,000 detailed requirements for ALWR designs. The NRC published their Safety Evaluation Report (SER) [2] in 1992 and 1994 detailing their review of the URD.

Work on the URD provided a foundation for follow-on-design programs, as well as the development of an integrated, industry-wide plan for and commitment to the resolution of challenges to building new nuclear power plants. The goal of the overall program is described in the Strategic Plan for Building New Nuclear Power Plants [3], published by the Nuclear Energy Institute (NEI) Executive Committee to resolve the full spectrum of issues, technical and institutional, in an integrated and coordinated way. The Strategic Plan was first published in 1990 and has been updated annually to reflect progress made, new challenges, and revised milestones to address these challenges.

The remaining sections of this paper describe the development, organization and content, and use of the URD in ALWR implementation. Much of the content of this paper is from Volume I of the URD.

2. DEVELOPMENT OF THE REQUIREMENTS DOCUMENT

Overall direction in developing the URD has been provided by the ALWR Utility Steering Committee (USC) consisting of senior representatives from approximately twenty U.S. and foreign utilities. EPRI personnel have acted as staff to the USC. EPRI also established contracts with U.S. nuclear steam supply system vendors, as well as engineering service, consulting, architect-engineer, and construction companies. As a result, the ALWR requirements are driven by utilities, but have also had the benefit of participation of a broad range of industry participants. Thus, the requirements are essentially a consensus of the industry as to those features that should be sought in the next generation of plants.

The USC established policy statements in key areas which are central to the achievement of program objectives and which have broad, fundamental influence on plant design requirements. These policy areas tend to be pervasive ones which the utility sponsors consider important to correcting issues (e.g., plant simplification) in existing plants or to be ones which explain fundamental ALWR guiding principles (e.g., use of proven technology). The policy statements are not considered design requirements by themselves, but rather influence or form the foundation for a set of requirements. The fourteen policy statements are discussed further in Section 4.

The main focus of the requirements and commensurate level of detail is on areas of improvement needed to achieve an excellent power plant. Considering the policy statements, as well as the scope and focus of the requirements, the URD was organized into volumes, based on plant designs, and chapters based on plant systems and topics.

The development of individual chapters followed an iterative, consensus building approach. Initial chapter drafts were developed by selected teams of engineers from ALWR contractors. Chapter Managers were selected to lead the development effort and consensus building process. Periodic meetings with representatives from contractor organizations were held to review and resolve comments on chapter drafts. A formal database of comments and their resolution supported the drafting process. Periodic meetings were held to apprise the USC of progress and key issues associated with the development of each chapter. When chapter drafts achieved sufficient maturity and all comments were resolved, they were provided to the USC for review and approval. In September 1990, after approximately five years of development, the first complete revision of the URD was sent to the NRC for their review.

The development of the URD was an unprecedented effort by utilities to provide, at new levels of comprehensiveness and clarity, their desires for future nuclear plant designs. The URD development process presented an opportunity for utilities to consider various approaches for effective communication with plant designers. As the process evolved, preferences regarding the format, level of detail, and style for writing requirements became clear, as did the need to tailor requirements based on the characteristics of the subject matter. The specificity and comprehensiveness employed for developing the requirements varied depending on the operational feedback and improvements desired. Considerable emphasis was placed on the incorporation of human factors in the design process and improvements in the man-machine interface in general, and specifically the main control room.

Once the first version of the URD was produced, a means of maintaining and updating the document was required. A formal process was established to request changes to the URD, to have changes reviewed by all interested participants, to obtain approval of changes by the USC, and finally, to implement changes into numerical revisions to the URD.

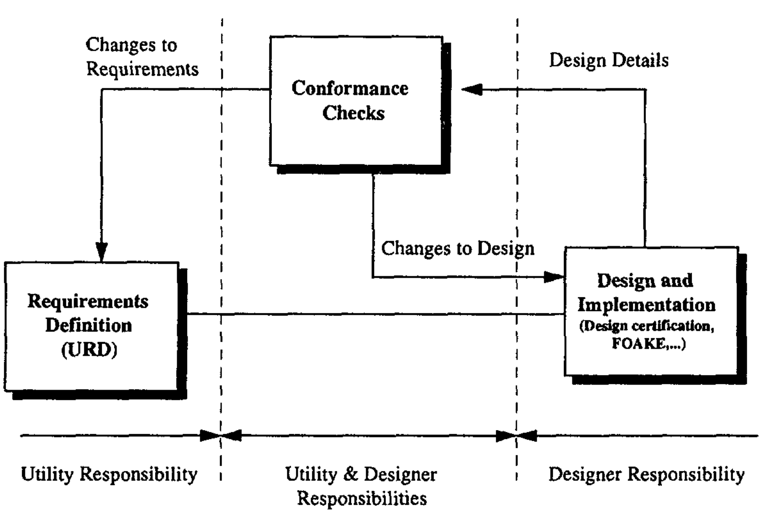

FIG. 1 ALWR Utility Requirements Document in the Design Process.

Changes to the requirements have been processed based on review comments from the NRC, and, more recently, on feedback from the ALWR design projects. For example, the latest revision of the URD has greatly benefited from the ABWR first-of-a-kind engineering (FOAKE) completed in 1996, and from the just completed AP600 design certification and FOAKE efforts. Figure 1 shows the model presently implemented between EPRI, utility representatives, and the designers for the development of the LWR designs. The feedback loops show that changes to the requirements, or to the design, may be needed to assure conformance between requirements and design.

To date, there have been seven revisions to the URD incorporating over 500 changes. The eighth revision incorporating another ~100 changes is planned for publication in June 1999. Given the sizable number of updates to the URD, it is clear that the URD has demonstrated its value in focusing and directing design processes toward the needs and desires of potential owners and operators.

3. ORGANIZATION OF THE REQUIREMENTS DOCUMENT

A systematic approach has been taken in developing and organizing the requirements. The URD covers the entire plant up to the grid interface. It therefore is the basis for an integrated plant design, i.e., nuclear steam supply system and balance of plant, and it emphasizes those areas that are most important to the objective of achieving an ALWR which is excellent with respect to safety, performance, constructibility, and economics. The URD applies to both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). An overall illustration of the structure of the document is provided in Figure 2.

Volume I of the document defines ALWR Program policy and summarizes top-tier design requirements. The policy statements provide utility positions on key aspects of design, development, and ALWR Program implementation. The top-tier design requirements are key elements in meeting ALWR Program objectives in order to make a viable nuclear power generation option available in the future. The top-tier design requirements also form the basis for developing detailed requirements in subsequent volumes for specific plant concepts.

Volume I is written in a narrative format (versus the requirements--rationale format used in Volumes II and III as described below) in order to present the policies and top-tier requirements in a more compact manner. Also included in Volume I is a section that defines ALWR cost goals to assure that the ALWR is economically competitive with other generation alternatives. Finally, a section is included on plausible scenarios for ALWR implementation, including certification, design, and construction.

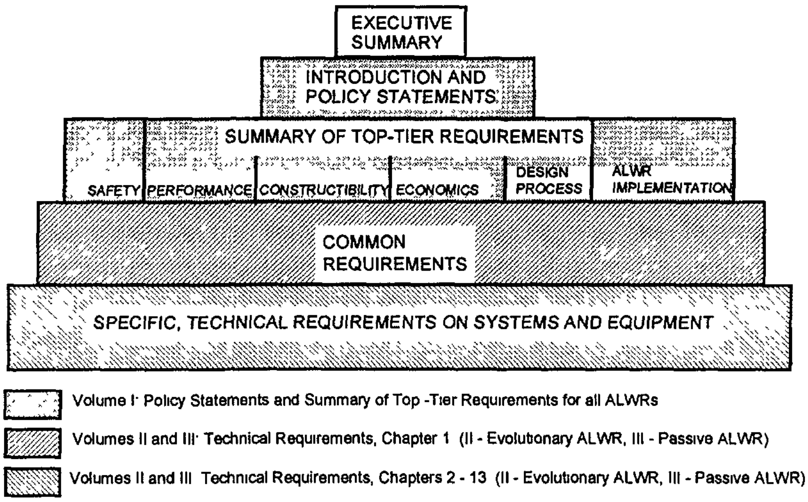

FIG. 2 ALWR Utility Requirements Document Organization.

Volumes II and III of the URD contain the complete set (top-tier and detailed) of design requirements for the Evolutionary and Passive ALWRs, respectively. Chapter 1 of each volume defines common requirements applicable to a number of plant systems. These requirements are in one chapter to avoid repetition in the subsequent chapters. Chapters 2 through 13 of each volume have been organized by groups of systems to cover the entire nuclear plant. URD chapter titles are shown in Table I.

3.1 Requirement - engineering rationale approach

The design requirements specified in the URD are organized in a side-by-side format that provides an engineering rationale for each requirement. The requirements define utility positions on the means for resolving issues in design, construction, and operation of current plants and for meeting the ALWR Program objectives. The rationale presents the basis for the requirement and provides later users of the document a better understanding of the requirement and its intent.

Volume I and the subsequent introductions to various sections of Volumes II and III include narrative text that is not in the side-by-side format. This narrative text typically states ALWR policy or necessary background. Although not strictly considered to be plant design requirements as is the side-by-side format, this narrative text provides the user with an understanding of ALWR policy, perspective on program background and the section scope.

3.2 Explanation of requirement terminology

The URD requirements are mandatory features and attributes of the ALWR design that are necessary to convince the Plant Owner that the plant will be excellent in all aspects. By definition then, requirements are directed at the plant design team, i.e., the Plant Designer, and compliance with

Table I. ALWR UTILITY REQUIREMENTS DOCUMENT CHAPTERS

Chapter	Title		Chapter	Title
1	Overall Requirements		5	Engineered Safety Systems
1A	PRA Key Assumptions and Groundrules		6	Building Design and Arrangement
1B	Licensing and Regulatory Requirements and Guidance		7	Fueling and Refueling
			8	Plant Cooling Water Systems
1C	Cost Estimating Groundrules		9	Site Support Systems
2	Power Generation Systems		10	Man-Machine Interface Systems
3	Reactor Coolant System and Reactor Non-Safety Auxiliary Systems		11	Electric Power Systems
			12	Radioactive Waste Processing Systems
4	Reactor Systems		13	Turbine Generator Systems

them should be demonstrable. Requirements are intended to be challenging, yet achievable. It is the intent of the ALWR Program to provide a set of compatible requirements that result in an integrated design that meets overall ALWR program objectives. The URD is not a set of requirements to be selected and chosen from, but rather to be invoked as an integrated set of requirements which establish the plant design basis for the Plant Designer.

There are a number of very desirable plant characteristics that are established as design requirements, but are in areas that pertain to factors beyond the Plant Designer's complete control, such as volume of radioactive waste produced, plant construction schedule, and plant availability. In these cases, the intent is to require the Plant Designer to develop and demonstrate a plant design for which the stated characteristic can be achieved by a competent and professional constructor and owner/ operator organization.

4. ALWR POLICIES AND TOP LEVEL REQUIREMENTS

The ALWR Program has formulated fourteen policies in order to provide guidance for overall URD development, and to provide guidance to the Plant Designer in applying the requirements. While not design requirements themselves, the policies cover fundamental ALWR principles that have a broad influence on the design requirements. Key policy statements are shown in Table II.

Table II. SUMMARY OF ALWR UTILITY REQUIREMENTS POLICIES

Simplification	Simplification is fundamental to ALWR success. Simplification opportunities are to be pursued with very high priority and assigned greater importance in design decisions than has been done in recent, operating plants; simplification is to be assessed primarily from the standpoint of the plant operator.
Design Margin	Like simplicity, design margin is of fundamental importance and is to be pursued with very high priority. It will be assigned greater importance in design decisions than has been done in recent, operating plants. Design margins which go beyond regulatory requirements are not to be traded off or eroded for regulatory purposes.
Human Factors	Human factors considerations incorporated into every step of the ALWR design process. Significant improvements will be made in the main control room design.
Safety	Excellence in safety for protection of the public, on-site personnel safety, and investment protection. Places primary emphasis on accident prevention as well as significant additional emphasis on mitigation. Containment performance during severe accidents is evaluated to assure that adequate containment margin exists.
Design Basis Versus Safety Margin	ALWR designs will include both safety design and safety margin requirements. Safety design requirements (Licensing Design Basis) are necessary to meet the NRC's regulations with conservative, licensing-based methods. Safety margin requirements (Safety Margin Basis) are Plant Owner-initiated features which address investment protection and severe accidents on a best estimate basis.
Regulatory Stabilization	ALWR Licensability is to be assured by resolving licensing issues, appropriately updating regulatory requirements, establishing acceptable severe accident provisions, and achieving a design consistent with regulatory requirements
Standardization	The ALWR utility requirements will form the technical foundation which leads the way to standardized, certified ALWR plant designs.

Table II. SUMMARY OF ALWR UTILITY REQUIREMENTS POLICIES (ctd)

Proven Technology	Employed throughout ALWR designs to minimize investment risk, control costs, take advantage of existing operating experience, and assure that a plant prototype is not required; proven technology is successful and clear demonstration in LWRs or other applicable industries such as fossil power and process industries.
Maintainability	Ease of maintenance to reduce operations and maintenance costs, reduce occupational exposure, and facilitate repair and replacement of equipment
Constructibility	The ALWR construction schedule will be substantially improved over existing plants and must provide a basis for investor confidence through use of a design-for-construction approach, and completed engineering prior to initiation of construction.
Quality Assurance	The responsibility for high quality design and construction work rests with the line management and personnel of the Plant Designer and Plant Constructor teams.
Economics	ALWR plants will be designed to have projected busbar costs with a sufficient cost advantage over competing baseload electricity generation technologies to offset higher capital investment risk associated with nuclear plant utilization.
Sabotage Protection	Inherent resistance to sabotage plus protection by plant security and integration of plant arrangements and system configuration with plant security design.
Good Neighbor	The ALWR plant will be designed to be a good neighbor to its surrounding environment and population by minimizing radioactive and chemical releases.

4.1 ALWR top-tier design requirements

A brief summary of top-tier utility design requirements is provided in Table III; categorized by major functions, including safety and investment protection, performance, and design process and constructibility. There is also a set of general utility design requirements, such as simplification and proven technology, which apply broadly to the ALWR design, and a set of economic goals for the ALWR program. These requirements reflect the ALWR Program policies described above and form the basis for developing detailed system design requirements for specific ALWR concepts.

Table III. SUMMARY OF TOP-TIER ALWR PLANT DESIGN REQUIREMENTS

GENERAL UTILITY DESIGN REQUIREMENTS	
Plant type and size	PWR or BWR, applicable to a range of sizes up to 1350 MWe • Reference size for Evolutionary ALWR: 1200-1300 MWe per unit • Reference size for Passive ALWR: 600 MWe per unit
Safety system concept	• Evolutionary ALWR - simplified, improved active systems • Passive ALWR - passive systems; no safety-related bulk ac power
Plant design life	60 years
Design philosophy	Simple, rugged, high design margin, based on proven technology; no power plant prototype required.
Plant siting envelope	Most available sites in U.S.; 0.3g Safe Shutdown Earthquake (SSE)
SAFETY AND INVESTMENT PROTECTION	
Accident resistance	Design features to minimize initiating event occurrence and severity: • Fuel thermal margin $\geq$ 15% • Slower plant response to upset conditions through features such as increased coolant inventory.
Core damage prevention	Design features to prevent initiating events from evolving to core damage.
• Core damage frequency	Demonstrate by PRA that core damage frequency $<10^{-5}$ per reactor year.
• LOCA protection	No fuel damage for up to a 6-inch break
• Station blackout coping	8 hours minimum coping time for core cooling (indefinite for Passive ALWR)
Mitigation	
• Severe accident risk	PRA whole body dose less than 25 rem at the site boundary for severe accidents with cumulative frequency greater than 10^{-6} per reactor year.
• Containment Design	Large, rugged containment building with design pressure based on Licensing Design Basis pipe break.
• Containment Margin	Margin in containment design is sufficient to maintain containment integrity and low leakage during severe accident.
• Licensing source term	Similar in concept to existing Reg. Guide TID-14844 approach, but more technically correct release fractions, release timing, and chemical form.

Table III. SUMMARY OF TOP-TIER ALWR PLANT DESIGN REQUIREMENTS (ct'd)

PERFORMANCE	
Design availability	87%
Refueling interval	24-month capability
Unplanned auto scrams	Less than 1/year
Maneuvering	Daily load follow
Load rejection	Loss of load without reactor or turbine trip for PWR (BWR from 40% power).
Operability and maintainability	
• Design for operation	Operability features designed into plant, such as: forgiving plant response for operators, design margin, and operator environment
• Design for maintenance	Maintainability features designed-in, such as: standardization of components, equipment design for minimal maintenance needs, provision of adequate access, improved working conditions , and ready access to equipment.
• Equipment replacement	Facilitate replacement of components, including steam generators.
Man-Machine Interface	
• I&C systems	Advanced technology, including software based systems, multiplexing, alarm prioritization, fault tolerance, and automatic testing.
• Operations simplicity	Single operator able to control plant during normal power operation.
DESIGN PROCESS AND CONSTRUCTIBILITY	
Total time from owner commitment to construct to commercial operation	1300 MWe evolutionary plant designed for 72 months or less 600 MWe passive plant designed for 60 months or less
Design status at time of initiation of construction	90% complete
Design and plan for construction	Design for simplicity and modularization to facilitate construction; develop an integrated construction plan through Plant Owner acceptance.
Design process	
• Design integration	Manage and execute design as a single, integrated process.
• Configuration control	Comprehensive system to control plant design basis.
• Information management	Computerized system to generate and utilize integrated plant information during design, construction, and operation

5. ROLE OF REQUIREMENTS IN ALWR IMPLEMENTATION

There are three primary roles of the ALWR design requirements in ALWR implementation as illustrated in Figure 3. The influence of the ALWR requirements is expected to pervade the entire ALWR implementation process.

5.1 Establishing a stabilized regulatory basis

The ALWR requirements have established a stabilized regulatory basis through actions in four areas: (1) licensing issue resolution, (2) regulatory requirement optimization, (3) establishing acceptable severe accident provisions, and (4) achieving a design consistent with regulatory criteria.

The key function of ALWR requirements here was to obtain meaningful agreements with the NRC, as reflected in the SER, in these four areas.

5.2 Providing requirements for certification design

The second primary role of the ALWR requirements is to provide a set of standardized technical requirements to be met by suppliers in their certification designs. It is in the vendors' interest to meet the ALWR requirements, because of the stabilized regulatory basis established by the require-

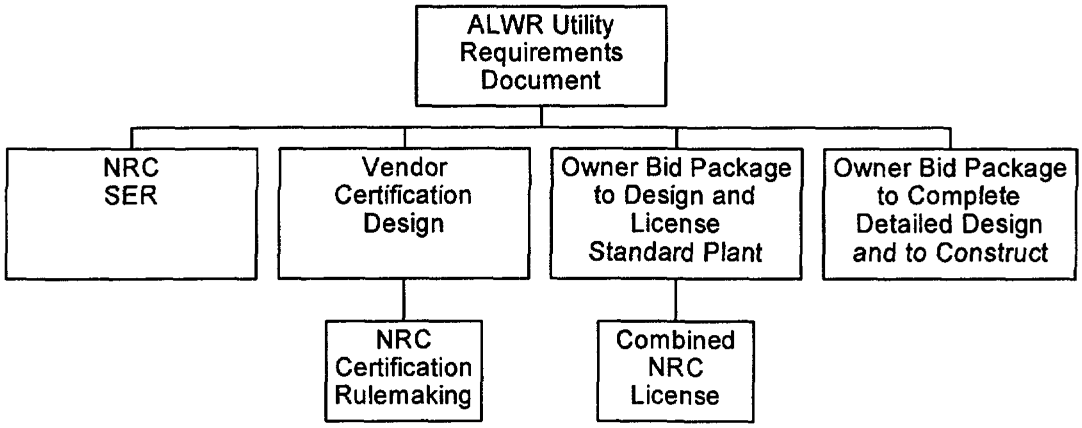

FIG. 3. Role of Requirements in ALWR Implementation.

ments and the fact that the requirements reflect the needs and desires of the electric utility industry. This utility industry has the plant operating experience and is likely to be a key participant in any ALWR investment group.

5.3 Providing requirements for owner bid packages

The third primary role of the ALWR requirements is to serve as the technical requirements for ALWR owner bid packages to design and license the standard plant. It is expected that any ALWR investor will insist on having an investment-ready design with high assurance of licensability. The ALWR requirements provide the foundation for this assurance. Also, the ALWR requirements will be an input to the owner bid package to complete the detailed design and to construct.

6. CONCLUSION

The URD is an important element in the process to develop new LWR designs for use in the twenty-first century. The policies and requirements embodied in the URD reflect the large volume of experience accumulated in operating LWR designs, and therefore provide a strong foundation for proceeding with confidence to develop and implement ALWRs that will meet the future needs of electrical energy suppliers.

The viability and utility of spending the time and resources to develop a set of utility requirements has been demonstrated in the early stages of the ALWR design process, and will become increasingly evident as designs are completed and plants are constructed and operated. As organizations in many countries around the world take the initiative to develop requirements specific to their needs, it has become clear that the development of these requirements has become an essential first step in the process for perspective owners and operators planning for the future implementation of nuclear power plants.

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FEEDBACK OF OPERATION AND MAINTENANCE EXPERIENCE INTO EVOLUTIONARY PLANT DESIGNS (HWRs)

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Abstract

The process of feeding back operation and maintenance information into the CANDU plant design process has been in constant evolution since the beginning of the CANDU program. The commissioning and operation experience from the first commercial reactors at Pickering A and Bruce A was used extensively in the design of the first generation CANDU 6 Plants. These units have been in operation for 15 years, producing electricity at an average lifetime capacity factor of about 85%. In further advancing the CANDU 6 & 9 design, greater emphasis is placed on enhancements that can reduce operational costs and further improve plant performance by reducing the planned outage time. The plant design has been improved to facilitate maintenance scheduling, equipment isolation, maintenance and post maintenance testing. Individual tasks have been analyzed as well as the interaction between tasks during outages to reduce the down time required and simplify the execution of the work. This results in shorter outages, reduced radioactive dose and reduced costs. The Utilities have continued to play an important role in CANDU 6 Evolution. Specifically; the Korea Utility KEPCO has one of the original four CANDU 6 Plants and three of the most modern. Their feedback to the designers has been very helpful. One of the most important feedback processes is through the CANDU Owners Group, which provides information exchange between members. In India eight PHWRs of 220 MWe capacity are in operation. Four reactors, also of 220 MWe capacity are in advanced stages of construction. Site construction work of two units of 500 MWe PHWRs at Tarapur will be taken up shortly. Over the years, during construction and operation of these power stations, a large amount of experience has been accumulated. Operation and maintenance experience is shared with operating stations by intensive participation of design engineers in Station Operation Review meetings, trouble shooting and root cause analysis of problems. Feedback from COG and WANO reports is also available. All this experience has been appropriately ploughed back into design of 500 MWe PHWR. This paper discusses the feedback process and some specific features of the current CANDU and other PHWR designs.

1. INTRODUCTION:

The process of feeding back operational and maintenance information into the CANDU plant design process has been in constant evolution since the beginning of the CANDU program. In the early years, the focus at AECL much like other nuclear plant suppliers was on improving the design from a technology perspective. The commissioning and operation experience from the first commercial reactors at Pickering A and Bruce A was used extensively in the design of the first generation CANDU 6 plants. These units have been in operation for 15 years, producing reliable energy with public risks well within the standards set by the regulatory agencies.

The strategy adopted by AECL has been to progressively evolve the CANDU design based on lessons learned from the operating CANDU's world-wide and other feedback sources (Figure 1). The units being built and recently placed in service in Wolsong, Korea and Qinshan, China are benefiting from these improvements and subsequent plants will benefit from further enhancements.

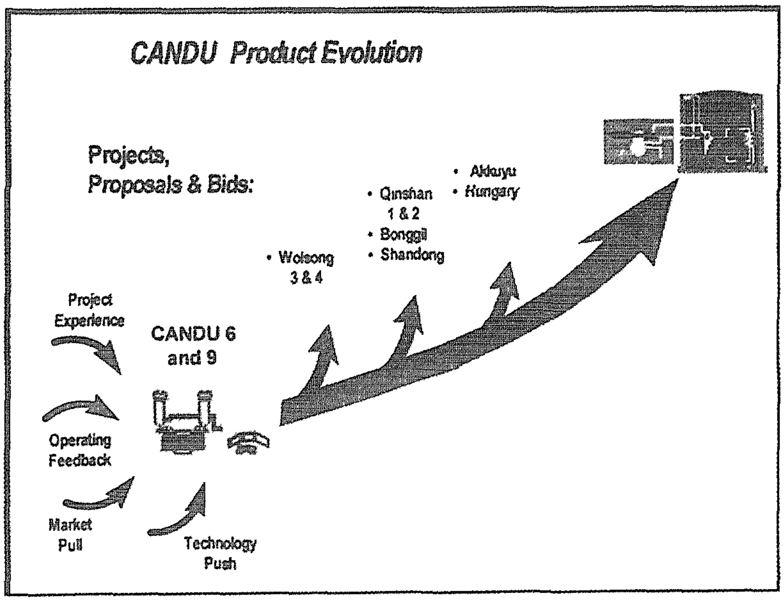

FIG. 1 CANDU Product evolution.

AECL is using a number of methods and processes to obtain feedback. One of the most important processes is through the CANDU Owners Group which provides information exchange between members. AECL also has its own internal operating experience feedback process which ensures that the lessons are systematically considered. AECL uses experienced utility staff in design reviews and in developing new products and services. AECL staff are also gaining valuable plant experience in plant commissioning, and in servicing operating CANDU plants. This ensures that the benefits from the feedback are realized for both future customers as well as existing plant operators. The CANDU design has long been recognized for its robust and reliable technology. The design continues to evolve and is a very competitive alternative for electricity generation into the twenty first century.

2. OPERATION AND MAINTENANCE FEEDBACK SYSTEM

AECL has implemented a formal process of gathering and responding to feedback from operation, construction, commissioning, and from regulatory activities (Figure 2). A design feedback

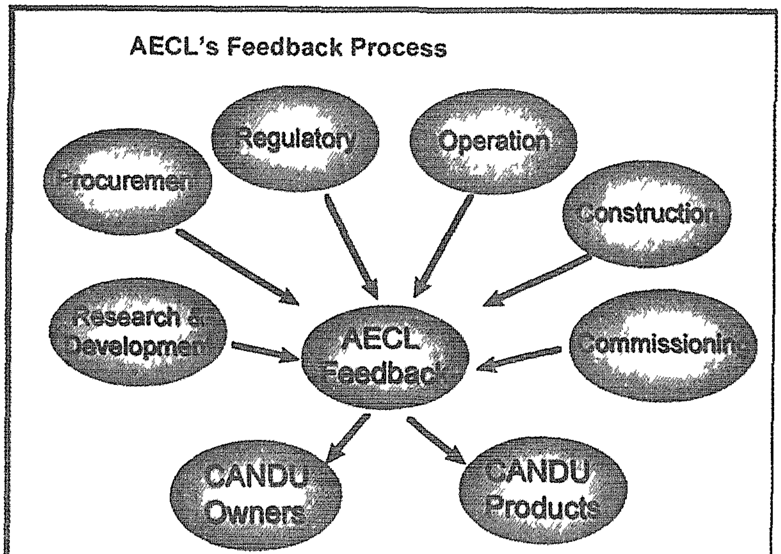

FIG 2 The AECL Feedback process.

team ensures that all feedback is entered in the CANDU feedback database for follow-up. This CANDU feedback database is available to all design staff, on-line via the AECL Intranet.

Feedback from CANDU plant operations, including operating events from light water reactors, is increasingly being evaluated and collected in a comprehensive and systematic manner. Lessons learned are used as input to both the design process and the design features of the CANDU products. This build-up of knowledge and insight from operating experience ensures that AECL is more effective in responding to future and current stations issues.

Recent examples of operational input to designs are:

- ECC performance improvements including strainer design for long term circulation of emergency coolant;
- Improvements to liquid relief valve component and system design for heat transport pressure and inventory control;
- Improved safety system signal monitoring with updated digital displays, increased computerized testing for reduced operation burden and increased system reliability;
- Steam generator operational improvements such as provision of additional access ports for chemical cleaning and improved divider plates for long term performance;
- For the CANDU stations now under construction the Control Centre design and technology has been upgraded to improve safety and capacity factors by reducing the probability of human error.

3. PLIM (PLANT LIFE MANAGEMENT) & PLEX (PLANT LIFE EXTENSION)

Today, customers are seeking a safe and reliable technology which can generate electricity at a competitive price during the entire plant design life. Provisions for life attainment and extension are also extremely important to maximize the return on initial investment. Building in knowledge from operations and maintenance feedback regarding equipment aging, is a crucial element of a successful PLIM program. AECL is working with CANDU utilities on a PLIM program for operating CANDU plants.

The overall objectives of the PLIM /PLEX program are:

- To perform a comprehensive assessment of the critical systems, structures and components and develop or enhance the plant inspection, maintenance and rehabilitation programs to effectively manage the effects of aging.
- To ensure continuing safe, reliable and cost effective operation of existing CANDU stations in accordance with the following goals:
  a) That the risk to the public from operation of CANDU units is well within the regulatory requirements throughout the nominal design life of 40 years* (life assurance);
  b) An achievable 85% capacity factor lifetime, provides electricity at a competitive cost during the nominal design life of 40 years (life assurance);
  c) Ensure major unexpected problems are avoided through identification of potential aging issues before their occurrence. Means for monitoring and mitigation to ensure reliable component performance are implemented in a timely fashion.
  d) That the life extension option beyond the nominal design life of 40 years is preserved.

The PLIM/PLEX program is an ideal example of design/operations/maintenance co-operation (Figure 3).

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* Many of the early plants had a specified design life of 30 years.

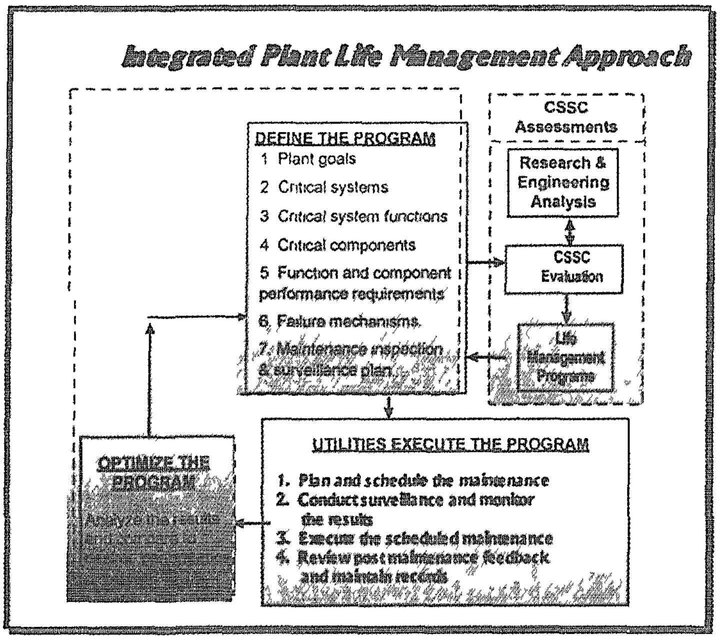

FIG. 3 Integrated Plant Life Management.

Initially, the designers define the critical systems, structures and components ("CSSC"). Comprehensive assessment of the current status and subsequent identification of aging mechanisms leads to the definition of maintenance and surveillance plans. Next the utilities execute these maintenance and surveillance activities and feedback the results to the designers.

Designers then analyze the results and compare them to desired performance. Finally, recommendations for further inspections and/or modified operating conditions are incorporated into the PLIM program and transmitted to the utilities.

A recent example of this closed loop PLIM process was the feeder wall thinning experienced at several CANDU Stations. As a result of the program operating chemistry was changed and enhanced inspections specified. Material changes were incorporated into new designs.

Another key element of the PLIM/PLEX program is Reliability Centred Maintenance (RCM). A comprehensive systems assessment is carried out using a process that allows for importance ranking as well as risk ranking thereby providing an overall criticality ranking.

- For each system, functions important to plant safety, environment and reliability are identified.
- For each function, the failure mechanisms and their impact on plant goals are identified. In particular, the requirements established in the plant safety analysis and probabilistic safety analysis are reviewed in detail to ensure that any failure mechanisms which could impact on the analytical assumptions are identified.
- Components which could cause these failures are then identified. Both active and passive components are included.
- Failure modes and effect analysis for these components are then completed. For components where the failure modes and effect analysis cannot be completed due to lack of information, then these components are added to the list for special aging studies.

This approach provides a comprehensive Reliability Centred Maintenance program that goes far beyond the list of safety or economically important CSSCs and ensures that the plant surveillance, inspection and maintenance program are enhanced to cater to aging mechanisms before they impact on plant safety or performance.

4. INFORMATION TECHNOLOGY IMPROVEMENTS

The information technology advances that have revolutionized the business world have also created opportunities to improve operations and safety for future CANDU, and will provide the vehicle for reducing operation and maintenance costs.

The next generation CANDU control centre will provide improved safety, reduced maintenance costs, a reduction in forced outages, obsolescence protection and the opportunity for a reduced station staff count (Figure 4). This control centre has been designed making use of operations and maintenance feedback as an integral part of the design process. For example, the design of the advanced alarm management system has been based on extensive review with plant control room staff, regarding their experience in interpreting the large number of annunciation messages during plant upsets.

The design provides the capability to integrate displays, alarm messages, interactive operator to plant communication sequences, procedure support and surveillance scenarios.

The underlying computer information systems are flexible, expandable and easy to upgrade to exploit evolving technology. The design provides for a substantial reduction in operational complexity and equipment diversity. This leads directly to reduced maintenance costs. The incorporation of most of the detailed hard-wired analog and digital relay controls into a single distributed control system substantially reduces the component count, diversity and complexity of the instrumentation, wiring and controls. For example the number of wire terminations related to the plant controls has been reduced from about 80,000 to 25,000. The same technology can provide CANDU utilities with instantaneous two-way plant wide and corporate wide real time plant information communications capability. Personnel in the plant's chemistry lab, administration building, maintenance shops and executive offices can all have access to the same real time information as the operators in the control room. This technology has been designed to provide for substantial obsolescence protection. The hardware and system software components are all proven off-the-shelf items integrated to take advantage of the most relevant published and defacto international standards. For example IBM PC (Intel) computers and the VME electronic packaging standards.

The CANDU control centre design, incorporating the full set of advancements to the man-machine interface and the above advanced computer systems, is being proof-tested in a mock-up at AECL. This mock-up includes detailed event simulations. In addition, these advanced features are field tested at the operating stations to gain operations feedback.

5. USE OF ADVANCED ENGINEERING TOOLS

The design engineering for Qinshan CANDU 6 project and the basic engineering program for CANDU 9 utilizes advanced engineering tools such as 3-Dimensional Computer Aided Design (CADDS) tools, intelligent flow sheets, automated wiring, cabling and detailed instrument and control logic design, automated pipe support design/analysis and linked equipment and material management databases, to achieve a reduced delivery schedule and better design and construction quality. The 3-D CADDS model is used to decide the layout configuration and space allocations; to optimize the fabrication construction sequences; and to determine the use of prefabricated assemblies depending on the layout and complexities of the systems. The ability to thoroughly check the design, before construction starts, using powerful interference checking routines will avoid costly rework and delays. The enhanced presentation of the construction packages by CADDS will improve information transfer to construction staff.

For project management of the Qinshan project, AECL has also implemented a new computerized engineering design and document management system to keep track of the engineering outputs and deliverables to suppliers and site construction groups, as well as project correspondence. The drawings for the construction of the Qinshan project are also managed by the same system. This is the first time that the construction aspect of a CANDU project has been computerized using AIM and TRAK application software. AIM is an electronic document storage system and TRAK, developed by AECL, is an electronic document package production system.

Having made these advances in the design/construction phase of a project the next logical step is to utilize the systems in the operation and maintenance of a plant.

In recent years, plant configuration management has become an area of attention for plant operators worldwide. This can be a costly program if not managed early in plant life through effective processes. The systems described above will greatly simplify operations configuration management tasks.

6. COOPERATIVE PROGRAMS

AECL has been active in the CANDU Owners' Group Information Exchange program where good operating practices are exchanged, events are reported, and data relevant to safe plant operation is collected screened, and distributed to COG members. AECL and the CANDU Utilities have also held a number of annual Operating Experience Feedback Seminars to allow further exchange between utility operators and the designers. In recent years, there have been a total of five COG/IAEA Technical Committee meetings on Exchange of Operational Safety Experience of PHWRs where all PHWR utilities world-wide can share and exchange their operational experience and solutions to safety issues.

7. EVOLUTION IN THE KOREAN HWR PROGRAM

KEPCO, the Korean electrical utility, is aware of the value of providing operational feedback to the designer in order to ensure a gainful evolution of the design takes place. For this reason, KEPCO has always cooperated with AECL to provide information related to operability, maintainability, and cost of all elements of the plant.

The benefits of providing feedback to a responsive designer were evident, from looking at the design enhancements incorporated into the CANDU 6 design between the Pickering A and Bruce A designs and that of Wolsong-1. This has resulted in a unit with a lifetime performance record, which stands very high on the list of world achievers.

When KEPCO made the decision to build three more CANDU units (Wolsong-2/3/4), they gave serious thought to how they could make use of the experience gained through their years of operation of their first CANDU 6 to gain even further advantage from the proven robust design. After considerable discussion with AECL, agreement was reached on the incorporation of 74 design changes in the Process & Equipment, Instrumentation & Control, and Fuel Handling areas. These changes were achieved while maintaining the concept that the operational interfaces for Wolsong-2/3/4 should be maintained as close as possible to those of Wolsong-1.

At present, KEPCO is considering all available options for the expansion of their nuclear power generation capacity. A logical part of this consideration is how any known operational difficulties and maintenance problems which might exist in the previous station can be identified and eliminated in the future design. KEPCO recognizes that, if the process for identifying and incorporating items where improvement would be desirable is to work, it must involve an effort both on the part of the utility and of the designer.

The Wolsong-2/3/4 project has been effective in facilitating the passage of information from the plant to the designer in the following ways:

- All field design changes performed at site have been forwarded by the site organization back to the designer for information and/or action.

- Korean involvement in the CANDU design had been ensured by assigning system design authority to an established Korean design organization (KOPEC).
- Although the project was executed in Korea almost entirely with Korean staff, a small support staff of expatriates, both at site and at the KOPEC design office, were witness to the requirements for enhancement or change, where they existed, and were personally able to provide feedback to their parent organizations.
- KEPCO has always shown cooperation in providing feedback information to AECL, even when there is no direct or immediate benefit to them in doing so.
- KEPCO is actively involved in seeking out areas where improvements would be beneficial, and in meeting with AECL staff to keep them informed.

As a utility responsible for producing electrical power safely and economically, KEPCO realizes that it is to their benefit to provide information to the designer openly and in detail, when that information can be used to improve not only the existing plants but also future designs of those plants.

8. EVOLUTION IN THE INDIAN HWR PROGRAM

India launched its Pressurized Heavy Water Reactor (PHWR) programme with Rajasthan Atomic Power Station (RAPS), the design of which was prepared by AECL, Canada based on the Douglas Point Reactor. At present eight PHWRs of 220 MWe capacity are in operation. Four reactors, also of 220 MWe capacity are in advanced stages of construction. Site construction work of two units of 500 MWe PHWR at Tarapur has been recently started. The plants subsequent to RAPS have incorporated considerable technological changes and upgrading of the design. The design changes have been motivated by a variety of considerations which include feedback from construction, operation and maintenance and international experiences, evolution in safety and regulatory practices and a need to synergise designs with the capabilities of Indian industrial infrastructure.

Over the years, during construction and operation of these power stations, a large amount of experience has been accumulated. Operation and maintenance experience is shared with operating stations by intensive participation of design engineers in Station Operation Review meetings, trouble shooting and root cause analysis of problems. Valuable experience is gained while participating in meetings of the Safety Review Committee for Operating Plants (SARCOP), the Project Design Safety Committee (PDSC) and also IAEA sponsored specialists' meetings and programmes. Feedback from COG and WANO reports are also available. All this experience has been appropriately ploughed back into the design of the 500 MWe PHWR.

The evolutionary design concepts that have been adopted for PHWRs in India are generally similar to those in CANDU designs elsewhere. In this regard, as examples, we could cite incorporation of two independent diverse fast acting shutdown systems, high pressure emergency core cooling system, independent safety trains, use of Zr-Nb coolant tubes, four tight-fitting garter springs, annulus gas monitoring system, calandria - end shield assembly, reduction of number of valves in PHT system, layout changes to facilitate operation maintenance and reduction of manrem, control room designs with computer based operator information system, computer data logging to facilitate event analysis, supplementary control room etc.

There are certain systems or approaches which have evolved in the Indian context in a different direction compared to CANDU. As an illustration of such areas of evolutionary designs, two examples are described below:

8.1 Fuel Handling System

Functionally, on-power refuelling in bi-directional mode from the upstream end of the channel has been retained in all Indian PHWRs as was in RAPS. As the designs were evolved for Narora

Atomic Power Station (NAPS), the 220 MWe standardised design, the following changes were incorporated.

a) The pressure vessel of Fuelling Machine Head (magazine housing) was changed from forged and welded construction to forged and bolted type. Guide sleeve latching was changed from hydraulic to mechanical type.

b) The floor-mounted cart-type Fuelling Machine carriage moving on horizontal rails was changed to a vertical bridge column carriage assembly.

c) The well proven 'Shuttle Transport System' to transfer a pair of fuel bundles contained in a shuttle assembly to the spent fuel storage bay has been retained. However, the rest of the fuel transfer system has been totally redesigned. Two independent Fuel Transfer Systems, one corresponding to each Fuelling Machine vault, have been provided. This feature also provides a certain amount of redundancy to continue refuelling, in an emergency, even while a section of the fuel transfer equipment is out of service. Two transfer magazines incorporated in these fuel transfer systems provide a closed heavy water environment in which all the spent fuel from a fuelling machine is temporarily stored, releasing the fuelling machine for fuelling the next channel. The subsequent operations for discharging the spent fuel into the Spent Fuel Storage Bay simultaneously with the fuelling of the next channel are performed as two independent threads. Thus the overall cycle time for refuelling a channel has been reduced.

Evolution of Fuel Handling System for 500 MWe PHWR is through significant modifications to the NAPS systems. In the 500 MWe Fuelling Machine (FM) head, the ram assemblies are powered by rack and pinion drives using oil hydraulic motors. However, an emergency backup using heavy water pressure of the ram housing has also been provided. The emphasis in detailed design of the FM head is on simple, robust and modular sub-systems for ease of maintenance. A full fledged, off-line, Calibration and Maintenance Facility (CMF) has been provided for these fuelling machines. This facility has the necessary features for carrying out maintenance, testing and calibration of important components, critical sub-assemblies and also complete FM head testing with simulated reactor conditions outside the reactor building in a tritium-free environment. The FM heads can easily be wheeled from the reactor into the CMF, which is located in the service building.

8.2 Containment System

Current Indian PHWRs use a double containment principle. The annular space between the primary and secondary containment envelopes is provided with a purging arrangement to maintain a negative pressure in the space. This arrangement significantly reduces the ground level releases to the environment during accidents when there is a release of radioactivity into the primary containment.

Both inner and outer containment structures are of concrete. The primary containment is the inner structure, consisting of a cylinder wall topped by a dome, both of pre-stressed concrete. The outer, or secondary, containment envelope is a reinforced concrete cylindrical wall topped by a reinforced concrete dome. The primary containment uses epoxy coating as liner for added leak-tightness and ease of decontamination. An important feature of the containment buildings is that sealed openings are engineered into the domes, to enable the steam generators to be lifted out vertically (using an external mobile crane {Figure 5}), should such a need arise, at a future date, as a part of SG replacement programme.

A pressure suppression system incorporating a suppression pool is used for limiting the peak pressure in the containment following a loss-of-coolant accident or a Main Steam Line Break (MSLB). The primary containment building is divided into two volumes, Volume V1 and Volume V2. Volume V1 encompasses the portion of building where high enthalpy section of Primary Heat Transport System is placed. The rest of the primary containment is Volume V2. These two volumes are separated by leak-tight walls and floors and are connected through a vent system via a suppression pool containing water in the sub-basement. During a LOCA or MSLB, the pressure rise in Volume V1

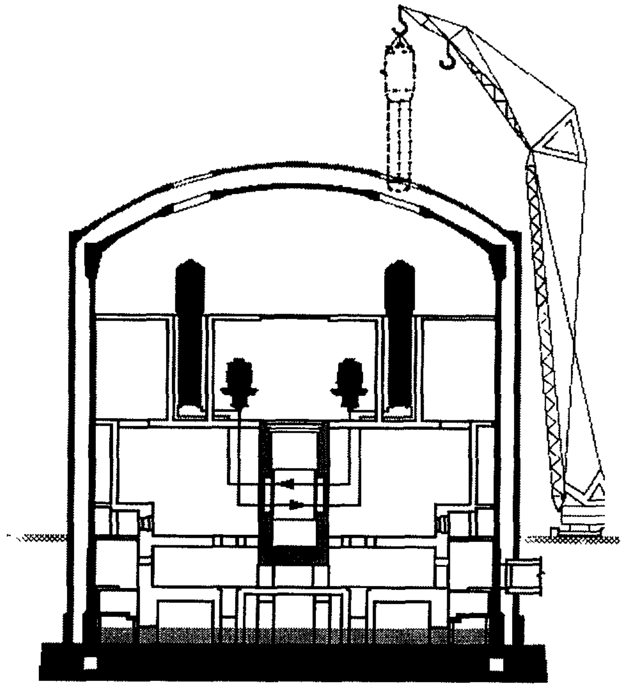

FIG. 5 Containment of Indian PHWRs.

will cause steam air mixture to flow via the vent system to the suppression pool, where the steam will condense, and the air will escape into Volume V2.

In addition to pressure suppression, the suppression pool water forms part of the long-term recirculation mode of emergency core cooling.

In the 220 MWe PHWR, the primary containment is designed for an internal pressure of 1.73 kg/sq.cm(g) based on peak pressure arising from MSLB, while for the secondary containment, the design pressure is 0.07 kg/sq.cm(g). For the 500 MWe PHWR, the corresponding pressures are 1.44 kg/sq.cm(g) and 0.13 kg/sq.cm(g) respectively.

For post-accident clean-up of the atmosphere in the containment, two systems are used:

a) Primary containment filtration and pump-back system: In this system, air flow is recirculated within the primary containment through charcoal filters, to perform containment atmosphere cleanup operation on a long-term basis after an accident.

b) Secondary containment filtration, recirculation and purge system: This system provides multi-pass filtration and mixing by recirculation within the secondary containment space and also maintains negative pressure within it. The negative pressure maintained in the secondary containment space brings the net ground level release down to very low values.

9. SUMMARY

The CANDU and Indian HWR programs have both extensively utilized feedback from operations and maintenance to improve their current products.

There are many examples, which show the benefit of this team approach.

As a result of these enhancements, plant life attainment and extension are possible; higher capacity factors will be achieved; operating costs reduced; maintenance focused on areas of significance; and finally operating issues such as configuration management can be cost effectively managed.

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OPTIMIZATION OF THE FUEL CYCLE

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Abstract

The nuclear fuel cycle can be optimized subject to a wide range of criteria. Prime amongst these are economics, sustainability of resources, environmental aspects, and proliferation-resistance of the fuel cycle. Other specific national objectives will also be important . These criteria, and their relative importance, will vary from country to country, and with time. There is no single fuel cycle strategy that is optimal for all countries. Within the short term, the industry is attached to dominant thermal reactor technologies , which themselves have two main variants, a cycle closed by reprocessing of spent fuel and subsequent recycling and a once through one where spent fuel is stored in advance of geological disposal. However, even with current technologies, much can be done to optimize the fuel cycles to meet the relevant criteria. In the long term, resource sustainability can be assured for centuries through the use of fast breeder reactors, supporting high-conversion thermal reactors, possibly also utilizing the thorium cycle. These must, however, meet the other key criteria by being both economic and safe.

1. INTRODUCTION

Nuclear power has to demonstrate its worth to a critical world and it is recognized that the industry must maximize its performance in order to satisfy its critics. This requirement leads to the concept of optimizing the fuel cycle, which must be carefully defined.

We must firstly recognize that the nuclear industry has a very long time horizon. In the shorter term, which in nuclear terms extends to decades, the industry is attached to a predominant technology, that of uranium-powered thermal reactors. The key variants are reactor type (LWRs and PHWRs), and whether the spent fuel is recycled to close the cycle, or stored with the intention of final disposal in a deep underground repository. Given known current plans, development of these reactors is likely to be evolutionary rather than revolutionary and such fuel cycles are likely to remain important in fifty years time. Nevertheless, there are many options for optimizing these cycles, using existing reactors and fuel cycle technologies. Moreover, further opportunities for optimizing the fuel cycle in this time frame may arise from developments in fuel cycle technologies, such as enrichment, new advanced recycling processes, remote handling and fuel fabrication, reactor operations, advanced fuels, and waste management.

Over the longer term, the possibilities for optimizing the fuel cycle are potentially more extensive. Most of these options involve a much greater use of the resource base for nuclear power and typically involve fast breeder reactors and the introduction of new reactor fuels such as thorium. However, the caveat is that fuel cycle decisions taken today can impact the options available in the longer term. There is a finite resource of fissile material that can be used to initiate fast reactors or thorium fuel cycles. The rate of introduction of these new fuel cycles, and their ability to meet energy requirements in the future, will depend on the extent and availability of fissile material when it is needed. Fuel cycle decisions taken today, can either open or close doors in the future.

Optimization itself is not a value-free term. What is an optimal solution for one party may not be so for another. In particular, different countries may simultaneously have very different energy policy requirements, based on many factors, including access to, and diversity, cost and security of energy resources; the state of industrial development; availability and cost of fuel cycle technologies, both domestically and off-shore (such as enrichment and reprocessing); back-end considerations, including total inventories of spent fuel and high-level waste; and government policy on energy and industrial development. We must also recognize that over time, concepts of what is optimal may alter substantially, according to what are regarded as the relevant criteria. This is highly relevant in the case of nuclear power, as it is such a long-term business. Reactors take several years to plan and construct, but also then operate for 40 years and beyond. Given the pace of change in the modern world, it seems highly likely that what is regarded as an optimal solution during the development of a reactor program will not necessarily prove to be the case throughout the whole lifetime of a plant. Changes to operating practices may sometimes be made along the way, but the world can frequently become stuck with choices which eventually prove to be sub-optimal.

Given the historical difficulty in predicting the availability and cost of energy resources and fuel cycle technologies, and the large uncertainties and variability in many of the factors, a superior nuclear energy strategy must include fuel cycle flexibility. This will enable a country, or utility, to optimize its fuel cycle strategy based on its own unique circumstances, and to change that strategy when those change.

It is also important to note that the fuel cycle must be seen as a whole in the discussion of optimization. It is possible to seek best solutions for each stage or element in isolation, without considering the effect on the whole. This is poor practice, given the important inter-relationships between the stages, where the knock-on effects of choices have to be evaluated. This "whole" includes consideration of not only the entire fuel cycle, but also of the entire national, or even regional energy environment. It is in this context that Korea, for instance, is considering exploiting the synergism between their PWRs and CANDU reactors through the DUPIC fuel cycle.

It is useful to firstly explore some of the possible concepts of optimization of the nuclear fuel cycle before briefly examining their application in a historical context. The analysis can then be brought up to date by examining how technology and practices can relate to the alternative concepts, in both the short and long terms, before reaching some conclusions on how the nuclear industry may proceed.

2. ALTERNATIVE CONCEPTS OF OPTIMIZATION

There are a wide variety of potential criteria for optimizing the nuclear fuel cycle. Some fuel cycle options may possibly optimize all of the criteria simultaneously; the more common situation involves a trade-off between conflicting criteria. The four most important criteria are arguably economics, sustainability and security of energy resources, environmental, and proliferation resistance, although this list is by no means exhaustive. Safety is not included in the list of potential criteria, as all fuel cycle options must meet the highest levels of safety, and there is usually little to differentiate between different fuel cycle options on the basis of safety.

For the electricity utility today operating an existing nuclear power plant, economic optimization means minimizing the costs of producing electricity, based on a high degree of plant reliability, operational flexibility and maximum fuel utilization. This will include the costs of spent fuel management and eventual plant decommissioning. The restructuring of electricity markets which is taking place throughout the world means that nuclear plants must compete directly with other modes of generation. Only if plants can prove cost effectiveness will they survive.

With regard to possible new plants, they must again prove their economic viability against their rivals. Although some countries still adhere to the concept of national energy planning, the invisible hand of the liberalized market is gradually taking over decision-making in the energy world. Optimization of the fuel cycle therefore fundamentally implies selecting options which constrain costs. Another important point is that the time horizon of private financial markets is relatively short, so there is an incentive to minimize costs which come early in a project and obtain a financial return as quickly as possible. Utilities are also very conservative, risk-adverse, and reluctant to make changes to proven methods unless the benefit is compelling. This favors evolutionary improvements to plant design, and well-proven fuel cycle options based on existing technologies and facilities, rather than attempting anything very innovative, which may experience cost over-runs and delays to the arrival of the revenue stream from electricity sales.

Even if the time horizon is opened up to allow fundamental technology shifts, there is likely to remain a strong economic criterion to be satisfied. Selected options will have to make economic sense, but considering a longer period allows the inclusion of a wider range of costs and benefits. For example, it is likely that external costs of energy supply programs will increasingly be incorporated in decisions, in particular the key environmental factors. In this regard, it is important to note that nuclear power internalizes the costs arising from ensuring safety and radioactive waste management and decommissioning of facilities. This means that these costs are included in the price of electricity generated by nuclear power. On the other hand, the costs arising from the adverse environmental and health impacts of other electricity generating options remain to be fully internalized. If this happens, it will serve to increase the cost competitiveness of nuclear power compared to fossil fuel burning for electricity generation.

The resource optimization criterion includes several factors: the cost, availability, and sustainability of the world's natural resources; local or national energy-resource independence and security of supply (whether natural or enriched uranium); diversity of energy resources (including non-nuclear). Some contend that nuclear power is now repeating the mistakes of the exploitation of other energy resources, such as coal, oil and gas, where market-controlled exploitation has taken place with little regard for its sustainability into the longer term. In an extreme view, optimization of the nuclear fuel cycle considering only the resource optimization criteria would only take place when the maximum amount of energy is extracted from each kilogram of uranium. Based on current operational fuel cycles, this is clearly not taking place, which implies that over the longer term, the process must have a strictly limited life. Uranium reserves may currently appear extensive, but an upsurge in nuclear power based on current fuel cycles may have a limited life of only fifty years or so.

Some adherents of the sustainability view of optimization would claim that theirs is the only credible longer term perspective and that it fits in well with the latest intellectual thinking about energy. For the developing countries in particular, it is clear that the pattern of energy exploitation followed the early industrialized countries cannot be repeated, as it is not sustainable in terms of known resources. It is also not sustainable in environmental terms, given concerns about the build-up of greenhouse gases in the atmosphere, due to the burning of fossil fuels. It is not seen as justifiable, however, to have the economic development of the world's poorer countries constrained by a problem which has arisen owing to misuse of resources by another set of countries. In addition, countries with relatively weak energy resource endowments feel the need to protect what they do possess, for fear of undermining the security of their economic development and of incurring substantial foreign exchange costs through energy imports. Recent international economic turmoil has provided support for this view. So far as nuclear power is concerned, it can offer the world certain advantages in environmental terms, but with a move to alternative fuel cycles in the longer term, may also be able to offer an unsurpassed level of sustainability, towards the level of those energy resources which are deemed renewable.

Nuclear fuel cycles have been subjected to a significant degree of outside scrutiny, which has produced a mass of information on the safety of reactors and other operating plants within the fuel cycle and the environmental aspects of spent fuel management. Evidence in these very difficult areas is, however, seldom clear-cut and decisions on what to prefer frequently require value judgments about what is inferior or superior. In these areas, it is notoriously difficult to be scientific, as so much is determined by people's perceptions of risk, which is often not based on objective reality.

Two examples illustrate the difficulty in establishing environmental criteria in an objective manner. One often-quoted waste management measure-of-merit is the volume of spent fuel or high level radioactive waste. In fact, it is well established that in a geological repository, such as that envisioned by Canada or Sweden, it is the decay-heat loading of the spent fuel or high level waste, rather than the volume, that is the main determinant in the size and cost of the repository. Hence, for example, the higher volume of spent natural uranium CANDU fuel is offset by its much lower decay heat, that allows a higher packing density in the repository compared to spent LWR fuel. As a consequence, the disposal costs and repository areas are similar for spent LWR and CANDU fuels. Another example is the focus on the radioactive source term, or the "radio-toxicity" of the spent fuel. Again, for geological disposal in a repository following the Canadian or Swedish concept, the contribution from the actinides in spent fuel to the long-term radiological dose is negligible. In a properly engineered repository, in a reducing environment, the actinides are virtually insoluble. Hence, one must be careful to consider the total risk and environmental impact of fuel cycle options, rather than focusing on only one narrow aspect.

The identification of non-proliferation optimization criteria is even more controversial. In the context of reactor and fuel cycle choices and future technological development in the civil nuclear power sector, the nuclear non-proliferation regime has been able to provide the necessary assurances, irrespective of the nuclear technology chosen, and should be able to do so in the future. In other words, nuclear fuel cycles and their facilities can be safeguarded. Nonetheless, concerns over proliferation aspects of fuel cycles has played a major role in shaping the nuclear industry today. The rejection of the reprocessing and recycling option in the United States is the most obvious illustration of this; and this policy continues to influence fuel cycle choices in other countries. On the other hand, the successful implementation of reprocessing and MOX fuel fabrication technology in Europe demonstrates that safeguards can be applied to fuel cycles involving the separation of fissile material to minimize proliferation risk. Controversy even extends today to spent fuel disposal, where some argue that once the spent fuel has cooled, either prior to or after final disposal, it may prove to be an easier source of plutonium for any party possessing the appropriate separation technology.

There are novel fuel cycles which contain a high level of proliferation resistance of which the DUPIC (Direct Use of Spent PWR fuel In CANDU) is an example. The high degree of proliferation resistance stems from several characteristics, including that there is no purposeful separation of isotopes (nor can the processes be easily tampered with to effect such a separation), the plutonium concentration is dilute (making it much more difficult for the removal of a significant quantity) and all stages of the process, as well as the final DUPIC fuel bundles, are highly radioactive (making physical access to the material, and its removal, extremely difficult). Between DUPIC and conventional reprocessing a range of other recycling technologies can be envisioned that have various degrees of proliferation resistance. The proliferation resistance of conventional reprocessing can be increased by not separating the plutonium and uranium to the same extent, and by leaving some fission products. In the so-called "TANDEM" cycle, uranium and plutonium from spent PWR fuel are co-precipitated for use as CANDU fuel; some fission products could be left as well to further enhance the proliferation resistance of this fuel cycle.

To understand the application of the various optimization criteria, it is worthwhile delving back briefly into history. Although the nuclear era is only half a century old, there have already been significant changes in perceptions of how nuclear power should develop, with the important lesson for us today that the future pattern is unlikely to be any different. Fuel cycle options which are readily criticized as irrelevant today may quickly become accepted once circumstances change.

3. SOME HISTORY

In the early days of the commercial nuclear power industry, it was always believed that owing to the scarcity of the world's uranium resources, the uranium-based fuel cycle would be a short-lived phenomenon. Optimization at that time therefore involved moving towards a fast-breeder reactor cycle to maximize utilization of scarce resources. This would initially involve the closing of the fuel cycle by the reprocessing of spent fuel to extract usable uranium and plutonium in order to commence the commercial development of fast breeder reactors. The use of MOX fuel in thermal reactors was considered as only an intermediate stage in the process. At the time, it was believed that the economic and sustainability concepts of fuel cycle optimization would coincide in the future, on the basis that the cheapest long term choice was also the sustainable one. The conception was that uranium would become very expensive as its scarcity increased, adversely affecting the economics of nuclear power.

There are many reasons why this has not happened, but in particular the much slower than expected growth of nuclear power and the discovery of extensive additional world uranium resources. Uranium today is no longer regarded as a scarce resource in most countries as there are abundant supplies available to fuel current nuclear technology for many decades. The introduction into the fuel cycle of uranium formerly tied up in nuclear weapons has strengthened this view.

Fast reactor development has also been much slower than was anticipated. Part of this may be attributed to lower perceived necessity for their introduction, but the technical and economic challenges have also proved demanding. The public hysteria which has been created concerning the use of plutonium as a fuel may also have had an impact.

The position today is that for many countries, there is a direct conflict between the economic and sustainability concepts of optimization. In many countries, the pressures to constrain costs of generation in increasingly competitive electricity markets imply that fuel cycle options which contradict the sustainability criterion will be selected. The time horizon taken by investors is a major constraint - independent power producers (IPPs) typically look ahead only a decade, thus either ruling out nuclear power or greatly limiting possible fuel cycle options.

There are exceptions to this. In India, with poor uranium resources and concerns about the energy security and foreign exchange costs of acquiring alternative supplies, a closed fuel cycle is seen as an essential step towards more resource-efficient fuel cycles based on fast breeder reactors, and high conversion heavy-water-moderated thermal reactors using thorium. The Russian Federation regard their uranium and plutonium resources as important national assets, irrespective of perceived abundance of uranium on world markets. They are resources not to be squandered by adopting options which may appear to be low cost with only a short term perspective.

Otherwise plants must prove their economic viability on a short term basis, or they are threatened with closure. Some of the solutions to achieving this may involve increasing the utilization of the uranium raw material through either high fuel burnups in LWRs or via reprocessing and eventual recycling of spent fuel. This will, however go only a small way towards hitting the sustainability criterion, as only a small part of the potential energy will be extracted from the raw material.

It is useful at this point to examine how technology and practices are seeking to address the various optimization criteria, initially in the shorter term when the current main fuel cycles are effectively fixed and then in the longer term, when there is a potential for a much greater range of solutions.

4. OPTIMIZATION IN THE SHORT TERM

As described in the previous section, the economic criterion is likely to dominate, subject to satisfaction of base requirements on safety, environment and non-proliferation. The prime means of achieving this will be to get reactors running as much as possible for as long as possible, in other words having high load factors over extended operating lives.

Considerable achievements in improving load factors have already been achieved with operating reactors, but there remain laggards performing well below the best in the industry. In LWRs, longer operating cycles, beyond the usual 12 months, are one route towards raising load factors, as are short, efficiently managed outage periods. Longer operating cycles do, however, have the disadvantage of increasing fuel cycle costs - it is always a matter of satisfying various conflicting objectives. Given the age structure of existing reactors, plant-life management and reactor refurbishment is becoming an increasing issue, involving in some cases, replacements of key components such as steam generators and, in others, the modernization of instrumentation and other components. Power uprating is also becoming increasingly common and this may greatly assist the economics of power production.

Economic optimization involves the minimization of all cost components. Operating and maintenance costs began to run out of control in some nuclear plants in the past, but are now strictly controlled. Another important area for attention is fuel costs and performance, with utilities tailoring their procurement practices to fit in with a perceived abundance of uranium and gradually moving towards higher fuel burnups for LWRs (which cut costs at both the front and back end). Fuel managers experiment with new fuel designs incorporating different enrichment levels in order to minimize costs for a given electricity output, with the trend towards higher average enrichment levels. This however, pushes LWR fuel technology towards its performance limits, which may bring some new technical problems. Back-end policies are still effectively decided on a national basis, but there are some signs that economic criteria will be more influential here when reprocessing contracts come up for renewal. Nuclear generators are increasingly having to sell their sole product in a competitive market and must seek low-cost solutions in every area.

Despite the low level of new plant orders in Western countries, considerable progress is still being achieved with evolutionary plant designs based on existing reactor types. Again the motivation is primarily economic, with a focus on nuclear power's relatively high capital cost, as nuclear power's previous financial advantage over similar base load electricity generation options has clearly been substantially eroded by low fossil fuel prices, and in particular, by increases in the thermal efficiency of the latest combined-cycle gas plants. The competitiveness of new nuclear plants can also benefit greatly from increased thermodynamic efficiencies, achieved through higher coolant temperatures. This in turn may require further advances in fuel technology. New designs aim to provide utilities with a high degree of fuel cycle flexibility, to fit in with their own unique circumstances Many aim at achieving economies of scale with unit sizes from 1000 to 1800 MWe - these will, however, only suit major electrical grids. It is also hoped that standardization of key nuclear plant designs will improve economics, as should the practice of having several reactors based on one site, where integrated operating and maintenance can be realized.

On the sustainability criterion, while present fuel cycles are clearly far from optimal, significant improvements in resource utilization can still be achieved using existing technology. Reprocessing and recycling of the recovered fissile material from LWR spent fuel back into LWRs can improve the overall uranium utilization by about 30%. Recycling the recovered uranium and plutonium from reprocessed spent LWR fuel into CANDU reactors would further improve the overall uranium utilization. Exploiting such fuel cycles and the natural synergism between LWRs and CANDU reactors can ensure sustainability for at least 50 years without a leap in technology (and perhaps much longer), until more resource-conserving fuel cycles are employed. The availability of environmentally friendly nuclear power is particularly important in developing countries, where much of the growth in energy and electricity demand is forecast in the next century.

Although future spent fuel reprocessing will become fundamentally an economic decision, there now exists new experimental technologies as an alternative to the PUREX process. Some hold out the possibility of being significantly cheaper, while not separating the uranium and plutonium to the same extent (which is attractive from the non-proliferation point of view). The DUPIC option that has previously been discussed in the context of its high degree of proliferation resistance, also has the potential of being simpler, and hence cheaper than conventional reprocessing.

There are some current fuel cycle features apart from recycling which also contribute to sustainability, albeit only slightly. Increased fuel burnups in LWRs economize on the uranium resource, while surplus world enrichment capacity is being used to re-enrich part of the enormous depleted uranium inventory and reconstitute it as low-enriched uranium (LEU). If the move to laser enrichment takes place, lower tails assays requiring a smaller uranium feed are likely.

With regard to the safety and environmental criterion for optimization, it may be accepted that there is little a priori reason to prefer either a once-through cycle or one closed by the recycling of reprocessed spent fuel, assuming that the facilities are operating properly. The arguments are quite finely balanced and hard to resolve in favor of one side or another. Moreover, from the resource utilization perspective, spent fuel disposal is not necessarily "final", as the public usually demands monitoring and retrievability. In an energy-scarce future, the fissile material in a repository could be recovered, if cost effective. What is certain is that there will be continued importance placed on safety and environmental considerations in an attempt to minimize the perception that these are key disadvantages of nuclear power. Those in the industry recognize that in fact, these are really the key advantages of the industry; the challenge is to convince the public that the industry is safe and environmentally sound, and that it should be the option of choice in meeting the challenges of global warming. The increased importance placed on safety and environmental impact will inevitably cost money, and although attempts will be made to reduce the regulatory burden where it is deemed inappropriate, it will remain as a significant cost. Some possible developments, such as higher fuel burnups for LWRs and lower enrichment plant tails assays, may reduce the amount of waste generated, but the prime motive will be economic. The key issue to be resolved for both the once-through and the recycle options, is convincing the public of the soundness of geological disposal for spent fuel and high level waste, without which confidence in the industry may diminish.

For non-proliferation criteria, shorter term optimization involves the application of strict safeguards throughout current fuel cycles. In reality, there is again little to choose between once-through and closed cycles. As noted above, advanced recycling technologies may be developed that not only reduce costs, but also reduce proliferation risk.

It is insightful to leave this discussion on the short-term optimization of the nuclear fuel cycle with an example that illustrates the simultaneous optimization of several criteria – the use of slightly enriched uranium (SEU) in CANDU reactors. The optimal enrichment that minimizes the fuel cycle costs in CANDU reactors is between 0.9% and 1.0%. Operational considerations may easily be met with enrichment at this level with no changes to the reactor, while fuel-cycle costs are reduced by 20 to 30% compared to natural uranium fueling. This cost saving is partly due to an improvement in uranium utilization (natural uranium requirements per unit of electricity generated are reduced by about 25%) and partly by spent fuel disposal costs being reduced relative to natural uranium by as much as 30%. SEU may also be used to up-rate the reactor power, which may provide a large economic benefit for both new and operating plants. Finally, the use of recycled uranium from reprocessed spent LWR fuel offers access to a potentially very economical supply of enrichment at the optimal enrichment level. Hence, SEU allows the optimization of several important criteria. The fact that it is not yet utilized by utilities underscores their conservative nature, which are generally focused on maintaining current plant operation

5. OPTIMIZATION IN THE LONG TERM

Optimizing the fuel cycle in the longer term essentially involves a significant improvement in the resource sustainability criterion which fails to be met by existing operational fuel cycles, together with satisfaction of the economic, safety, environmental and non-proliferation aspects. There are essentially two alternatives here, either the widespread introduction of technologies which make much more use of a scarce resource or an extension of fuel cycles to utilize new resources. The former is most likely to be satisfied by fast breeder reactors, while introduction of a thorium-based fuel cycle in current reactor types provides an example of the latter. However, as noted earlier, the caveat is that both cycles require fissile material for their initiation (and for fissile topping in the case of thorium cycles), and that resource is finite with reactor technologies currently in use. The rate of introduction of these new fuel cycles, and their ability to meet future energy requirements, will depend on the availability of fissile material when it is needed.

Despite good operational experiences with experimental and prototype fast reactors, their widespread introduction has been delayed and is now unlikely for several decades. The twin motives for their eventual introduction are likely to be the significant increase in the extraction of the potential energy from fuel together with the utilization of the plutonium and depleted uranium inventories which will have built up by that time. Owing to the breeding capability, the quantity of economically viable nuclear fuels may also increase significantly with fast reactors, taking in account currently uneconomic uranium deposits and also those of thorium.

One attractive option is a mixture of fast breeder reactors and lower-cost high-conversion thermal reactors utilizing thorium fuel, with the former supplying both initial and topping fissile requirements of the latter. While there is only limited knowledge of the extent of thorium resources in the world, the discovery of large deposits with high grade ores in several locations suggests that they are extensive. In principle, thorium-based fuels can be used in all current reactor types. Even though thorium does not contain a fissile component, ^{233}U is produced in-reactor through neutron capture in ^{232}Th, and subsequent beta-decay of ^{233}Th and ^{233}Pa. The concentration of ^{233}U in the spent fuel is about 5 times that of ^{239}Pu in spent natural uranium UO_2 fuel. This isotope of uranium is a very valuable fissile material because of the high number of neutrons produced per neutron absorbed (η) in a thermal neutron spectrum. The full exploitation of the energy potential from thorium requires recycling, which will not be economically justified for many years. Since commercial thorium fuel recycling facilities have not been built, there is an opportunity to develop a new, cheaper, proliferation-resistant technology for recycling.

This bridge between the thorium recycle options of the future and current uranium-based fuel cycles is the once-through thorium (OTT) cycle. This may prove attractive to countries having abundant thorium reserves, but are lacking in uranium. For example, two general approaches have been devised for OTT cycles in CANDU reactors. The first is a "mixed-core" approach, in which a large number of channels fueled with "driver" fuel would provide the external source of neutrons for a fewer number of channels fueled with ThO_2. This is the conventional OTT, and theoretically, values of enrichments, burnups, and relative feed rates can be chosen that may make this fuel cycle competitive (both in terms of resource utilization and economics) compared with natural uranium fuel. A "mixed-fuel bundle" approach is an alternative strategy for introducing the OTT cycle, which has the benefit of a particularly simple fuel management strategy. The "mixed-fuel bundle" contains ThO_2 in the central elements of a CANDU fuel bundle, and SEU in the outer 2 rings of elements.

Such a transition between the large installed thermal reactor system and one employing fast breeder reactors may take many years. The key resource sustainability and waste management advantages of FBRs will have to be balanced against their likely high initial capital costs but also their need for a demonstrated safety record. In bridging this gap between current and long term fuel cycles, policy makers will have to consider whether specific decisions preclude future fuel cycles options. One such fuel option that would close doors to future fuel cycle flexibility is the annihilation of plutonium (in for example, an inert matrix). While satisfying one fuel cycle criteria (proliferation resistance), this option would irrevocably reduce resource sustainability, by eliminating a future potential fissile material for initiating fast breeder or thorium fueled reactors.

6. CONCLUSIONS

In a climate where competition is gradually being introduced into electricity markets in many countries of the world and where there is an abundance of the uranium raw material, it is natural that the economic criterion is dominating moves towards optimizing the fuel cycle. It is essential that currently operating nuclear plants demonstrate their ability to prosper in this new environment. Some economic-based measures have either positive or negative implications for the other criteria but standards on safety, environment and non-proliferation must be enhanced rather than compromised. The next generation of evolutionary reactor designs should be able to make enhancements towards optimization on each of the criteria, but their introduction is heavily dependent on the current generation of plants demonstrating their worth. The incorporation of the advantageous environmental characteristics of nuclear power in future new generating capacity decisions would make new nuclear capacity more likely. The key to a shift to effectively infinitely resource sustainable fast breeder reactors is the acceptance of plutonium as a safe and economic fuel. This is still some way off, but is a reasonable objective for the industry in the medium term.

DESIGN DESCRIPTION PAPERS

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DESIGN DESCRIPTION OF THE EUROPEAN PRESSURIZED WATER REACTOR

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Abstract

The EPR (the European Pressurized Water Reactor) is an evolutionary PWR developed by Nuclear Power International and its parent companies, Framatome and Siemens, in co-operation with Electricité de France and German Utilities. NPI can rely on the huge experience gained by its parent companies; they have constructed more than 100 nuclear power plants throughout the world. The total installed capacity exceeds 100 000 MW - about 25% of the total world-wide figure. Following the conceptual design phase of the so-called Common Product conducted by NPI, Framatome and Siemens, from 1989 through 1991, Electricité de France (EdF) and several major German utilities decided to merge their own development programmes, - the N4 Plus and REP 2000 projects on the French side and the further development of the KONVOI technology on the German side, - with the NPI project. From that time on, the NPI project became one single common development line for both countries. In parallel, EDF and the German utilities decided to establish, together with other European utilities, specifications that would represent common utility views on the design and performance of future nuclear power plants. These are documented in the European Utility Requirements (EURs). The basic design has been completed in 1997, and in 1998 a design optimization is being carried out with the goal to even increase the economic competitiveness of nuclear power. This paper provides a brief design description of the EPR.

1. DESCRIPTION OF THE NUCLEAR SYSTEMS

1.1 Primary circuit and its main characteristics

The primary loop configuration is the same as that of existing designs and can be considered well proven.

The sizing of the reactor pressure vessel (RPV), steam generator (SG) (especially secondary side) and pressurizer (PZR) incorporates increases of the respective water volumes compared to current designs.

In the RPV design, the water volume between the level of the reactor coolant lines and the top of the active core is increased in order to improve the mitigation of LOCA (smaller breaks) by prolonging the period until beginning of core uncovery or minimizing the core uncovery depth, if any.

For the pressurizer a large volume is provided in order to smoothen plant response to operating transients and accidents.

The larger water volume of the SG secondary side supports smoothing of normal operating transients and reduces the potential for unplanned reactor trips. In case of a total loss of all feedwater supply (incl. emergency feedwater), the postulated dryout time of the SG will exceed 30 min.

The valve configuration of the primary side overpressure protection aims at avoiding the response of "non-isolatable valves" in plant conditions with a potential for radioactivity release. The valves are mounted to the top of the pressurizer in order to avoid high pressure piping. Each discharge train is provided with two safety valves in series; this tandem arrangement makes it possible to isolate a stuck-open safety valve without decreasing overpressure protection capability. Automatic opening of the main valves (of own medium operated type) is actuated by pilot actuators dedicated to each individual safety valve. During normal operation the valve assigned to the discharge function is closed, the valve assigned to the isolation function is open. For operation at lower temperatures, during stretch-out operation, the pilot actuators of at least one train are provided with remotely adjustable setpoints.

With the chosen arrangement of pressurizer discharge, the following safety functions can be performed:

- over-pressure protection of the reactor cooling system by automatically initiated discharge of either steam, water or two-phase fluid,

- depressurization of the reactor cooling system by discharge of steam, water or two-phase fluid in plant conditions when pressurizer spraying is not available or not efficient,
- discharge of the reactor cooling system to enable residual heat removal in case of unavailability of the secondary side heat removal (feed and bleed),
- discharge of the reactor cooling system in a postulated core melt situation to guarantee depressurization to a sufficiently low level that would rule out the high pressure core melt accident.

1.2 Reactor core and fuel design

The core is built up by 241 mechanically identically designed fuel assemblies; somewhat more than in currently operating units. Each fuel assembly consists of 264 fuel rods and 25 guide tubes arranged in a 17x17 array; their active length is 4.20 m.

The fuel rods are made of Zircaloy tubing containing uranium dioxide ceramic pellets, of which the initial enrichment is below or equal to 5.0 Wt %.

The average linear heat generation rate is about 178.6 W/cm, giving prospect of achieving average batch burnups of up to 65 GWd/tU. The design offers a high degree of flexibility with respect to cycle length adaptations, allowing fuel cycle cost reductions by high burnups and low leakage loading patterns.

Basic safety objectives are met by designing the core to have stabilizing reactivity coefficients under all operation conditions. Reactivity control is accomplished by changing the boron concentration in the primary coolant and by moving control assemblies. Slow reactivity changes caused by changes of xenon concentration and burnup are compensated by changes of the boron concentration, while fast reactivity changes for adaptation of the power level are compensated by control rod insertion or withdrawal.

The core is designed for UO_2 fuel assemblies and incorporates the capability also to insert MOX-fuel assemblies up to about 50%.

Some fuel assemblies contain burnable absorber (Gd_2O_3) to suppress high excess reactivity, especially in the first core. The reactor power level is monitored by the ex-core instrumentation. The capability to predict and to measure the three dimensional power distribution in the core is the duty of the in-core instrumentation system which consists of the aeroball system and the self-powered detector system. The in-core instrumentation penetrates the reactor pressure vessel head from the top at only very few positions. A cross-section of the core, showing the location of the core instrumentation, is depicted in Figure 1.

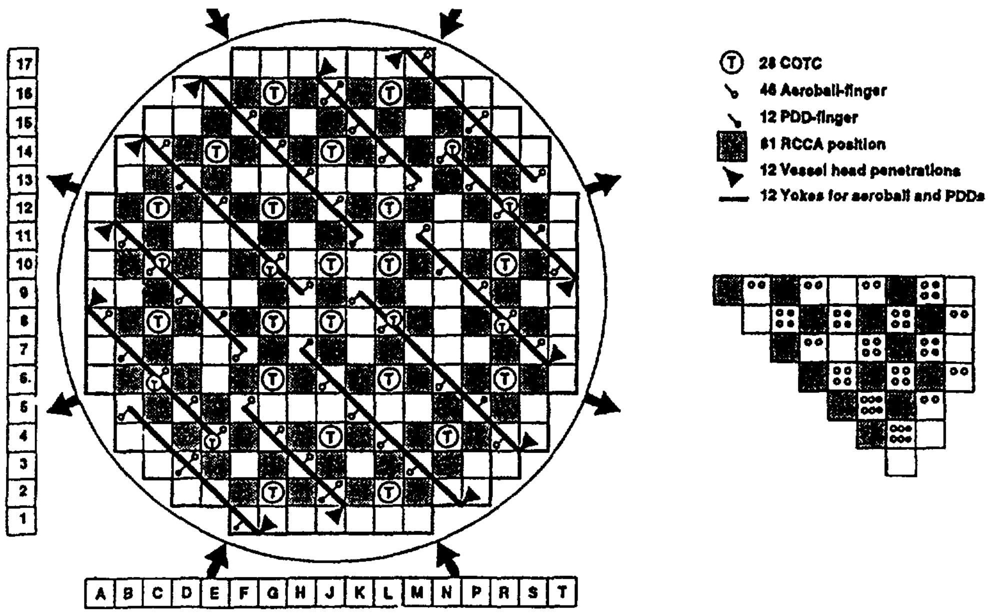

FIG. 1. EPR — Core instrumentation.

1.3 Primary components

1.3.1 Reactor pressure vessel

The reactor pressure vessel is designed for a life time of 60 years, not exceeding a total neutron fluence of 10^{19} nvt. This is achieved by provision of a rather large water gap and a heavy reflector around the core.

The upper part of the reactor pressure vessel will be machined out of one single forging. The flange is made as an integral part of the nozzle shell. The nozzles itself are of the on-set type so that non-destructive examination can be easily performed from the inside. The nozzles are located as high as practicable above the core upper edge to increase the hydrostatic pressure for reflooding and to avoid the loop seal effect.

The reactor pressure vessel is internally cladded with two layers of low carbon stainless steel. The inside surface is ground after cladding to a finish as required for ultrasonic inspection. Generally, in-service inspection is performed from inside the vessel; an access from the outside, between the outer wall and the thermal insulation, is also provided, however. Inspection from the outside will be performed, if the internal inspection should reveal indications which cannot be adequately characterized from the inside.

1.3.2 Steam generators

The steam generators feature an axial economizer to provide a steam pressure increase of about 3 bar when compared to a boiler type of the same heating surface.

The material for the tubes will be chosen from either Incoloy 800 or Inconel 690. Both materials have proven excellent properties regarding corrosion resistance and are exchangeable without affecting steam generator design parameters. The tubes are supported by perforated plates.

With respect to the pressure boundary, the same material as for the reactor pressure vessel will be chosen.

1.3.3 Pressurizer

The pressurizer is of conventional design but with an enlarged free volume. The spray systems for normal operation and auxiliary spray are completely separated from each other.

The spray lines are welded through a blind cover and equipped with a spray nozzle each. This design is easy to dismantle, inspect and replace. The spray system delivers a permanent flow to the spray nozzles to minimize thermal transients upon fast valve opening.

The heaters are flanged to the penetrations in order to be easily replaced and inspected.

All pressure boundary parts, except for the heater penetrations, are made of ferritic steel grade; basically the same as used for the reactor pressure vessel. The penetrations are in stainless steel and welded with an Inconel material.

1.3.4 Reactor coolant pumps

The reactor coolant pumps are of well-proven design, as already used in plants in France and Germany. The reactor coolant pumps are provided with a standstill seal in order to assure leak-tightness of the shaft seal without the need of an active seal water supply system under conditions when the pump is at rest, e.g. in the event of a station blackout.

1.3.5 Main coolant lines

With respect to the material of the main coolant lines, two options are still under consideration; either forged ferritic steel with austenitic cladding or forged stainless steel will be used. In any case, the break preclusion concept will be applied. A high quality in design, construction and surveillance enables preclusion of the current pipe break assumptions; consequently a catastrophic failure of a main coolant line is ruled out as regards its possible mechanical effects. However, a mass flow equivalent to a double area break of a main coolant line is still assumed for the design of e.g. the emergency core cooling system.

1.3.6 Reactor internals

The core barrel flange rests on a ledge machined from the flange of the reactor pressure vessel and is preloaded by an elastic system. The fuel assemblies rest on a flat perforated plate, machined from a forging of stainless steel and welded all around to the core barrel.

The cooling water flows through the core plate through four holes dedicated to each fuel assembly. These holes can be calibrated in such a way that a flat flow profile is achieved.

The space between the polygonal outside shape of the core and the cylindrical inner surface of the core barrel is filled by a stainless steel structure to reduce the fast neutron leakage and to flatten the power distribution. This structure is called the heavy reflector and represents an innovative feature compared to earlier designs.

The bulk of the internals is made of low carbon stainless steel in line with the current practice to prevent intergranular stress corrosion cracking in primary water environment.

1.4 Operating characteristics

The EPR is a nuclear island for a net electrical output of about 1750 MW. The primary components are enlarged relative to current types, and designs of the safety and operating systems have been updated accordingly.

The EPR is designed for being operated between 20 and 100% of rated generator power. In the power range between 50% and 100% load, the control systems will keep the average coolant temperature constant in accordance with the so-called part load diagram; the main steam pressure will vary between 8.4 and 7.25 MPa. In the lower power range, below 50% power, the main steam pressure is kept constant at 8.4 MPa and then the average coolant temperature will vary with the load. The advantage of this control strategy is that it results in the lowest demands on the chemical and volume control system, the loads on the pressurizer surge line and the control rod drive mechanisms during load changes in the most frequent operation mode.

The control and operational systems are designed to provide the EPR with a high capability to follow the actual power demands of the grid.

The load changes can either be initiated by the operator or completely remotely controlled. Important plant parameters are maintained within operational ranges automatically by control system functions, and the setpoints for the main NSSS controls are adjusted automatically; all plant parameters remain far from the triggering setpoints of any safety system during normal plant operation.

In addition, the EPR is designed to withstand without tripping of the reactor, events like: turbine trip, full load rejection, trip of one feedwater pump, and malfunction of a single control system.

2. INSTRUMENTATION AND CONTROL SYSTEMS

2.1 I & C Design concept, including control rooms

2.1.1 I&C structure

The functional requirements and failure models governing the design of the I&C systems are based on the overall safety criteria for system design and the functional requirements on the process systems. This implies a requirement for independent I&C subsystems in order to ensure that a loss of one subsystem will influence the remaining I&C systems only marginally.

The I&C systems and equipment are divided up into three classes (IE1, IE2, IE3) in accordance with their importance for safety and with respect to required reliability, performance, failure behaviour, maintenance, testing, and QA; in addition, there is a non-classified category (NC). The main features of these categories are:

- The I&C of the IE1 class that is used for automatic actuation and control of safety-grade systems, shall withstand a single failure also during periods of maintenance and testing. In addition, there must be no spurious actuation during maintenance or testing in case of a failure occurring anywhere upstream the last voter. This leads to a need for a fourfold redundant and divisionally separated structure.

- The I&C of the IE2 class, including information means and means for manual actions for mitigating measures during an accident, has a functional structure that can cope with a single failure - without the additional requirements related to maintenance and testing. Special emphasis is laid on the qualification of the back-up control and information means (safety control area), including the related software.

- The I&C of the IE3 and NC classes, that is used for normal operation of the plant, including the control of major plant parameters, limiting conditions of operation (LCOs), information and operation and other non-safety I&C functions, is specified in a case by case approach.

2.1.2 Applied technology

The proposed I&C automation and Man-Machine-Interface systems are based on utilisation of digital technology, preferably with "off-the-shelf" electronic components.

The potential for common cause failures is reduced by consequent use of functional diversity, for initiating parameters and actuation channels, and by distributing diverse I&C functions to I&C systems. Further, a formal specification of the I&C systems reduces failure potentials of software specification and makes the software easily verifiable.

2.1.3 Safety I&C (IE1, IE2)

The safety I&C functions shall have a high reliability so that they will not be a dominant contributor to the unavailability of safety systems.

The safety I&C is of redundant architecture, and designed to limit the consequences of equipment failures or malfunctions that may result from failure inducing events within I&C systems (single failure) and their consequential effects (with active or passive failure mode). Connections between redundant trains are necessary for exchange of information and commands, but they must not impair the independence. To this end, the redundant trains, or divisions, are installed with physical separation and with a minimum number of interconnections.

Interconnections are energetically decoupled against overvoltages from a disturbed division (e.g. by means of fibre optics), and erroneous signals from a disturbed division are prevented from affecting the other divisions by means of majority voting or signal coding. Necessary safety actions must be performed from the undisturbed divisions independent of the state of a disturbed division.

Appropriate measures are provided to cope with common cause failures (CCF) in order to meet the overall probabilistic design targets. CCFs and their consequences can result from different sources (e.g. faulty manufacturing, erroneous design, equipment failures, and environmental conditions during the course of accidents). CCFs are unavoidable, but the probability of their occurrence must be kept at a low level. Special probability values for digital I&C are practically nonexisting, and evaluations must therefore be based on engineering judgement.

The safety I&C functions, systems and associated equipment shall not be jeopardized by the operational I&C, and they are therefore decoupled from the operational I&C when interconnections can not be avoided. To this end, a "priority control" strategy is implemented; a safety command to an object used by both safety and operational I&C overrides any non-safety command.

2.1.4 Operational I&C

The operational I&C encompasses all I&C functions and associated systems and equipment for normal operation. It contains the measurements, the signal conditioning, open- and closed-loop controls, the signal processing and the data transfer to the man-machine interface.

The tasks of the open- and closed-loop controls are mainly to control the nuclear power generation during normal operation and anticipated operational occurrences in such a way that predetermined setpoints for relevant process variables are reached and maintained, to protect all mechanical equipment with high investment costs via redundant equipment protection, and to provide information for displaying the plant status for normal, upset and accident conditions and for documentation of all relevant process data.

Operational controls are operating in such a way that a sufficient margin to the actuation values of the safety I&C is maintained.

2.1.5 Man-machine interface facilities, and control rooms

The main control room (MCR) is a screen based control room with an overview panel. It is used for process control during normal, or accident situations including outages. In addition, the MCR has a safety control area with back-up control means. Further functions that are ensured from the main control room or from adjacent rooms are security surveillance, fire protection monitoring, radiation monitoring, management of maintenance and periodic testing, external and internal communication, access to documentation and to recorded information.

The MCR contains 3 operator work positions (all of the same design) which are used for process control in all plant conditions via operational I&C. A shift supervisor console offers operational and safety-qualified information to the shift supervisor, and/or to the safety engineer. It is equipped with communication means and space for administration work. The operator work positions are dedicated to the operators of the primary and secondary loops, and to the auxiliary operation or back-up purposes.

A plant overview panel is visible from all work places and will be used for the co-ordination among the operators and for the transfer between normal and back-up means.

The safety control area (with the back-up control means) in the MCR is used in the event of major losses of the normal control means. It can be used for the safe shutdown (hot or cold) of the plant or to perform post-accident operation. This area could also be screen based. The area constitutes a safety-relevant man-machine interface, and the related equipment is qualified accordingly.

The EPR is provided with a remote shutdown station (RSS) which is designed for transferring the plant to and maintaining it in safe shutdown conditions, in case of unavailability of the main control room without loss of operational or safety I&C systems. The RSS is equipped with internal and external communication means.

2.2 Reactor protection and other safety systems

The reactor protection system and other safety I&C systems are digital systems. They are characterized by divisional separation of their independent subsystems and the use of voting functions (2/4, 2/3, 2/4, etc.) for protective actions. The structure meets the requirements of the safety I&C functions described above.

3. SAFETY CONCEPT

3.1 Safety requirements and design philosophy

The strategy pursued for the EPR is to further enhance the already very high safety level attained at French and German plants. This strategy implies improving the prevention of accidents, including severe accidents, and adding features, mainly related to the containment, to mitigate the consequences of postulated severe accident scenarios - including core melt situations - to avoid need for stringent off-site countermeasures. The probability of such postulated accidents has been significantly reduced.

The plant design is based on a deterministic approach and considers "Risk Reduction" measures.

3.1.1 Deterministic design basis

In the deterministic analysis the different events are categorized in four Plant condition categories (PCCs) in accordance with their anticipated frequency of occurrence; PCC1 covers normal operation states, and PCC2 to PCC4 envelop transients, disturbed states and accidents.

Stringent radiological limits are applied for normal operation and anticipated operational occurrences as well as for accidents.

3.1.2 Risk reduction

The EPR design takes into consideration also events beyond the Deterministic design basis, - events with multiple failures and coincident occurrences up to the total loss of safety-grade systems, - on a probabilistic basis in order to quantify the residual risk. Severe accident design release limits have been specified on the basis that no stringent off-site emergency response actions (such as evacuation or resettlement) would be needed outside the immediate vicinity of the plant.

The overall safety objectives that have been set for the EPR, require that:

- the probability of core damage (CDF) shall be below 10^{-5}/year; and
- the probability of large releases to the environment shall be below 10^{-6}/year,

including all events and all reactor operating states.

In order to meet these objectives, some specific probabilistic design targets have been defined for the design phases:

- For internal events at power operation, the CDF shall be $< 10^{-6}$ per year;
- The contribution from shut-down states to the CDF for internal events shall be less than from the power states; and
- The CDF for internal events associated with early loss of containment function shall be below 10^{-7} per year.

Two Risk reduction categories (RRCs) have been introduced, and representative scenarios defined for both; - RRC-A relates to additional features to prevent accidents from progressing to a core melt situation, and RRC-B to prevention of large releases, - in order to provide a design basis for risk reduction features.

Typical examples on risk reduction features are:

- primary system discharge to the in-containment refuelling water storage tank, in the event of total loss of secondary side cooling (RRC-A); and
- features for spreading and cooling of corium, for hydrogen recombination, and for containment heat removal in the event of a core melt situation (RRC-B).

3.2 Safety systems and features

3.2.1 Safety systems configuration

Important safety systems are arranged in a four-train configuration as depicted in Figure 2.

The layout comprises four separate divisions, corresponding to the four trains. A simple and straight-forward system design approach is favoured, thereby facilitating operator understanding of plant response and minimizing configuration changes. The four-train configuration offers the possibility of extended periods of maintenance on parts or even entire systems, useful for preventive maintenance and repair work during normal operation.

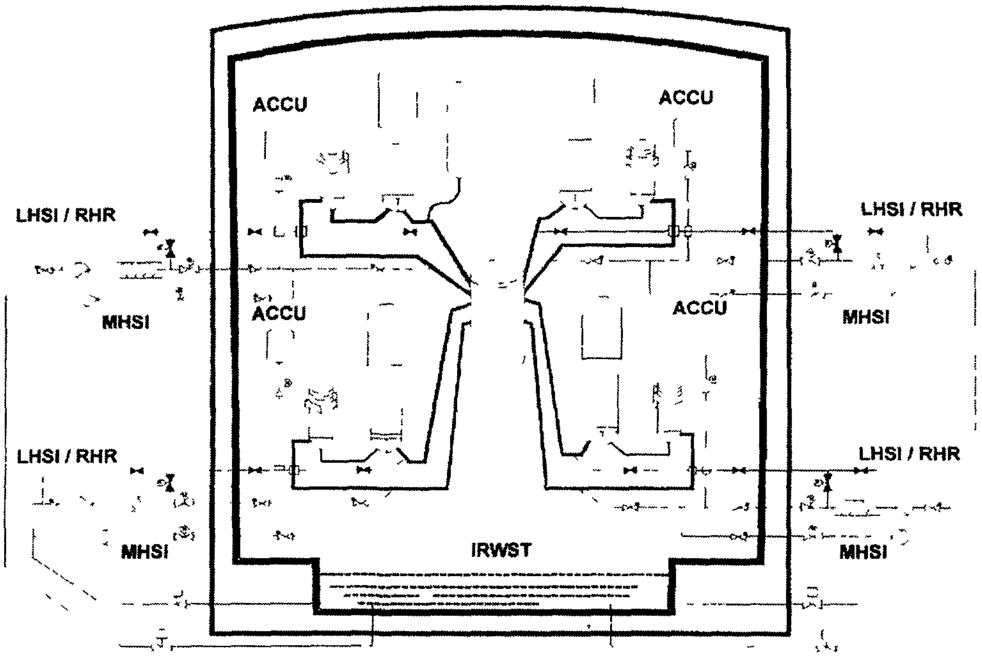

FIG. 2. EPR — Primary side safety systems.

The safety injection systems, for which an overview is presented in Table 1, feature an In-containment refuelling water storage tank (IRWST) located at the bottom of the containment. During design basis accidents the residual heat removal and low-head safety injection (LHSI) system transfers the decay heat to the ultimate heat sink via heat exchangers. The primary side safety systems are designed in accordance with stringent acceptance criteria to ensure limited fuel damages, even in case of large breaks. The delivery head of the medium head safety injection (MHSI) system will be adjusted below the steam generator relief and safety valve set points.

In case of a steam generator tube rupture, the affected steam generator will be isolated on the secondary side. After the initial transient, the primary and secondary pressures will equalize at a level below the set points of the safety valves in this steam generator, limiting to negligible levels the radiological releases.

The function of any one of the safety systems can be accomplished by another diverse system (or group of systems) in the event of malfunctions, as shown in Table 2.

In the EPR design efforts have been devoted to prevent high pressure core melt scenarios. Prevention of such scenarios implies the need for a highly reliable secondary side heat removal system.

Detailed investigations of active versus passive systems have led to the selection of an active emergency feedwater system with diversified power supply to the pumps to achieve a very high reliability. This system consists of four separate and independent trains, each with an emergency feedwater pump supplying feedwater to one of the four steam generators.

3.2 2 Safety injection systems

The safety injection systems mitigate loss of coolant accidents of all sizes, specific non-LOCA events, such as main steam line breaks and sequences leading to feed and bleed. The systems ensure heat removal, coolant inventory and reactivity control.

The medium head safety injection (MHSI) system feeds into the cold legs of the reactor coolant system. The shut-off head of the system is 8.0 MPa. This shut-off head is sufficient to cope with all LOCA related

TABLE 1. PRIMARY SIDE SAFETY SYSTEMS

MHSI Medium Head Safety Injection system	4 trains, cold side injection
Accumulators	4 accumulators, one per train, cold side injection
LHSI/RHRS Low Head Injection and Residual Heat Removal System	4 trains, combined hot and cold side injection in the long term
IRWST In-containment Refuelling Water Storage Tank	Storage of borated water inside containment

requirements, since a reliable secondary side partial cooldown is provided via safety-grade main steam relief valves. In conjunction with two small letdown lines, connecting two different hot legs with the IRWST, the MHSI system can be used for safety-grade boration during design basis accidents (PCC 2-4).

In addition to the medium head injection system, cold leg accumulator injection is provided to cope with large and intermediate break sizes. Four accumulators are provided, each directly assigned to one cold leg. The response pressure of the accumulators is designed to 4.5 MPa. The low pressure injection system with a shut-off head of 2.5 MPa together with the respective accumulator water volume provides a continuous water injection capability.

The low pressure injection system feeds initially into the cold leg. In order to stop the core outlet steaming and the steam release to the containment, a switching to combined injection into the hot and cold legs after 1-2 hours is foreseen. The injection pressure of 2.5 MPa offers advantages for feed and bleed operation and supports accumulator injection in an optimum way for a large spectrum of break sizes.

3.2.3 In-containment refuelling water storage tank

The In-containment refuelling water storage tank provides the source for emergency core cooling water and is located inside the containment between the reactor cavity and the missile protection cylinder on the

TABLE 2. DIVERSIFICATION OF SAFETY SYSTEMS

Complete failure of	Diverse system function		
MHSI	Fast secondary side pressure relief +	Accumulator injection +	LHSI Low head safety injection system
LHSI/RHRS (LOCA)	MHSI Medium head safety injection system +	Containment heat removal system (CHRS) (heat removal from IRWST) or	Secondary side heat removal via SG (small breaks)
LHSI/RHRS (Shutdown, RCS closed)	Secondary side heat removal via SG +	Steam generator feed systems	
LHSI/RHRS (Shutdown, RCS open)	Steaming into containment, CHRS if needed +	RCS make-up via MHSI	
Fuel pool cooling system	Fuel pool heat-up (boiling)	Coolant make-up via LHSI/RHRS	
Secondary side heat removal	Primary side bleed and feed		

bottom level of the containment. In the case of loss of coolant accidents, or in feed and bleed situations, the safety injection system draws from the In-containment refuelling water storage tank. The water steam mixture escaping through the leak, and through the bleed valve, respectively, is returned to the tank. In the case of severe accidents the In-containment water storage tank will provide the cooling water for flooding the spread molten corium.

In addition, the storage tank provides water for the operational function of flooding the reactor pit and the pools during refuelling.

3.2.4 Emergency feedwater system

The emergency feedwater system consists of four separate and independent trains. Each emergency feedwater pump takes suction from an emergency feedwater tank. These tanks and the systems are located in the four divisions of the safeguard buildings.

The EFW system does not have any operational functions. The four emergency feedwater pumps will be driven by electric motors which are emergency power supplied; in addition, two of them are connected to small diversified diesels so that the probability of common cause failure of all emergency power supplies is reduced to the minimum.

For start-up and shutdown a dedicated system is installed. This system is automatically started in case of loss of main feedwater and provides an efficient feature to minimize the need for the EFW system.

The emergency feedwater system transfers the residual and latent heat from the reactor coolant system via the steam generators to the atmosphere as long as the steam generator saturation temperature is above 150 °C, following any plant incident or accident other than those reactor coolant boundary ruptures for which complete residual heat removal by the safety injection system is possible (i.e. moderate to large size loss of coolant accidents). Following a LOCA in the size range which implies that not all core residual heat is released through the break flow and that at least a portion of the heat must be removed via the steam generators, the emergency feedwater system ensures sufficient water supplies to the steam generators.

In the case of a steam generator tube rupture, the emergency feedwater system removes the heat via the intact steam generators. The pressure in the affected steam generator is allowed to increase so as to reduce and eventually eliminate the break flow from primary to secondary side; the maximum pressure will remain at a level below the opening setpoint of the steam generator relief and safety valve.

The emergency feedwater system keeps the water inventory of at least one steam generator above an adequate level to maintain primary to secondary heat transfer, assuming a single failure.

Safety-grade, normally locked closed headers on pump discharge and on tank side ensure feed of all SGs and use of all water masses stored even in case of a single failure. These headers can only be opened after a sufficient grace period.

After a small break LOCA or a steam generator tube rupture, the emergency feedwater system provides enough cooldown capability so that the saturation pressure of the steam generator of 6.0 MPa is reached within a time span consistent with safety injection system performance requirements, and the radioactivity release limits for steam generator tube rupture, assuming a single failure.

The emergency feedwater system shall provide sufficient heat removal capacity and autonomy to ensure continued removal of decay heat for 24 hours with a final reactor coolant system temperature not exceeding nominal hot shutdown conditions. This shall be accomplished also under the assumption that no electric power is supplied from external sources and that the ultimate heat sink is not available.

3.2.5 Function of the combined low head injection and residual heat removal system (LHSI/RHRS)

The residual heat removal system is designed to transfer residual heat from the reactor coolant system via the cooling chain consisting of the component cooling water system and service water system to the

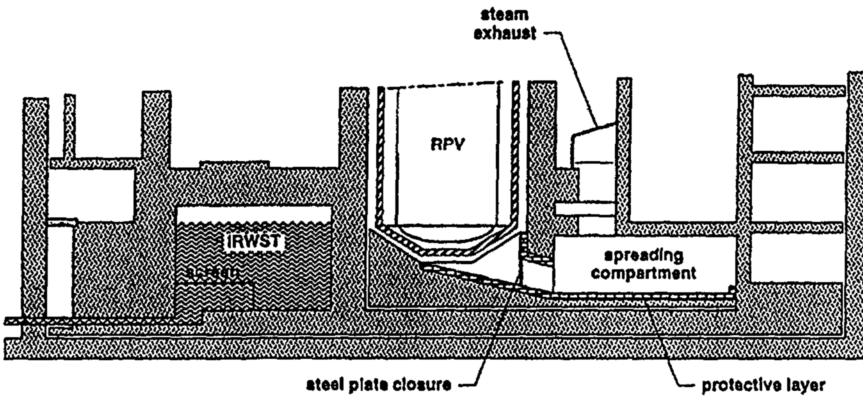

FIG. 3. EPR - Retention of molten corium.

ultimate heat sink, when heat removal via the steam generators is not sufficient. Furthermore, it ensures continued heat transfer from the reactor coolant system or from the In-containment refuelling water storage tank during cold shutdown or refuelling conditions.

The LHSI/RHRS combines operational and safety functions. The safety function implies that the residual heat removal system, in conjunction with the component cooling water system and the service water system, shall maintain the reactor coolant system core outlet and hot leg temperature below 180°C following a reactor shutdown, assuming a single failure and the maximum design temperature of the service water. The operational function requires the residual heat removal system to be capable of cooling down the reactor coolant system to 50°C following a reactor shutdown, with at least one reactor coolant pump in operation and considering the maximum design temperature of the service water.

The system consists of four separate and independent trains located outside the containment. Each of the four pumps draws water from a separate line connected to a hot leg of the reactor coolant system. The pump discharge is routed via heat exchangers to the cold legs of the reactor coolant system. A bypass line of the heat exchanger is provided to allow control of the cool-down rate. The residual heat removal system heat exchangers are cooled by the component cooling water system train, which is located in the same division as the associated residual heat removal train. Switch-over from secondary side cooling to residual heat removal cooling is foreseen at an average reactor coolant system temperature between 150 and 180°C.

3.3 Severe accidents (beyond design basis accidents)

3.3.1 Severe accident mitigation strategy

The EPR design enhances accident prevention and reduces the residual risk for the public and the environment by reducing possible releases of radioactive material and radiological consequences arising from severe core damage sequences

The design target of the EPR is that off-site emergency response actions (population evacuation or relocation) shall be restricted to the nearby plant vicinity. To this end, maintaining the integrity of the containment is highly important, and will be achieved by:

- Avoidance of early containment failure or bypass.
- Cooling of the corium in the containment and retention of fission products by water covering.
- Preservation of containment functions, such as low leak rates, reliable containment isolation function and prevention of basemat melt-through, ultimate pressure resistance to cope with energetic events.
- Pressure reduction inside the containment by dedicated heat removal
- Collection of unavoidable containment leakages in the annulus atmosphere and release via the stack after filtration.

3.3.2 Severe accident prevention and mitigation features

The EPR strategy includes both preventive measures and mitigating features:

- prevention of high pressure core melt situations, by ensuring a high reliability of the decay heat removal systems, complemented by pressurizer relief valves. The depressurization eliminates high pressure failure of the RPV and the danger of direct containment heating. The consequences of an instantaneous full cross-section break of the RPV at a pressure of about 2.0 MPa are nevertheless taken into account for the layout and support design.
- reduction of the hydrogen-concentration in the containment by catalytic H_2 -recombiners and, if necessary, by selectively arranged igniters. The prevention of molten core-concrete interaction contributes to reducing the amount of hydrogen.
- prevention of ex-vessel steam explosions endangering the containment integrity by minimizing the amount of water in the area where the corium is spread.
- prevention of a molten core-concrete interaction by spreading the corium in a spreading compartment provided with a protective layer (Figure 3), and
- connection of this spreading compartment to the reactor pit via a melt discharge channel which slopes towards the spreading compartment. This channel is closed by a steel plate, which will resist melt-through for a certain time, in order to accumulate the melt in the pit.
- provisions for connecting the spreading compartment with the In-containment refuelling water storage tank (IRWST) for water flooding after spreading; these pipe connections are closed during normal operation.
- dedicated basemat cooling system supplied by the containment heat removal system to prevent high temperature loadings in the basemat.

4. PLANT LAYOUT

The plant layout is governed by a number of principles derived from the huge experience gained through the construction and operation of the French and German nuclear power programmes with an installed capacity of more than 100 000 MW. The proven design contributes significantly to the economic viability of the nuclear power plant.

4.1 Buildings and structures, including plot plan

The general layout of the EPR plant is shown on Figure 4. The reactor building with the containment is surrounded by the safeguard and fuel buildings that contain the safety systems. The safety-grade systems are designed with a four-fold redundancy, arranged in four independent divisions with complete physical separation. Each division comprises a combined low head injection and residual heat removal system with the related intermediate cooling system, a medium head injection system and an emergency feedwater system.

The related electrical systems as well as the instrumentation and control systems are also allocated to these divisions but on a higher building level.

Other plant buildings, such as the access building and the nuclear auxiliary building, are located in close contact with the safeguard and fuel buildings, whereas the turbine building and the associated conventional electrical building are built separated from the reactor building complex and arranged so that the reactor building is located in the projection of the turbine generator shaft.

4.1.1 Design requirements

The plant is designed to withstand the impacts of internal and external events. With respect to earthquake and explosion pressure waves, the buildings and structures have been strengthened so that the function of safety-grade equipment will not be jeopardized by collapsing structures and that the equipment itself must withstand the dynamic effects inside the buildings. For protection against airplane crash, safety-related equipment is located in bunkers, or redundant portions will be geographically separated from each other so that only one train need be considered as impacted.

Aircraft crash: The assumptions with respect to aircraft crash are based on a probabilistic risk assessment, since statistical data are sufficiently representative and possible events are quite well known. The design load case for an aircraft crash has been determined on the basis of stipulations by the safety authorities.

Earthquake: Provisionally, the seismic design is based on the spectrum defined in the EUR (European Utility Requirements), scaled to 0.25g, for the free field level of horizontal movement, for a wide range of soil conditions.

Explosion pressure wave: A review of probability data regarding the risk of an impact by an external explosion for various sites indicates that they are closely related to the industrial environment of each site. For the EPR, the design is based on an incoming pressure wave with a maximum over-pressure of 10 kPa (100 mbar). The level of protection will be verified on a site by site basis.

The protection against external and internal hazards includes the divisional separation of safety-grade systems and the physical protection of the containment enclosing the reactor coolant pressure boundary By these means, the risk of inadmissible releases or common-mode failures of safety-grade system will be consistent with the deterministic design basis and the probabilistic targets of the EPR.

4.2 Reactor building

The reactor building (Figure 5) is the central building of the plant complex. In essence, it coincides completely with the containment, and thus, the following description of the containment covers also the reactor building.

4.3 Containment

The EPR has a double concrete containment design. The particular design concept uses, for the inner containment wall, the prestressed concrete technology. The leak-tightness requirement of less than 1 % volume per day can be ensured without provision of a containment liner. The outer wall, in reinforced concrete, completes the double containment arrangement.

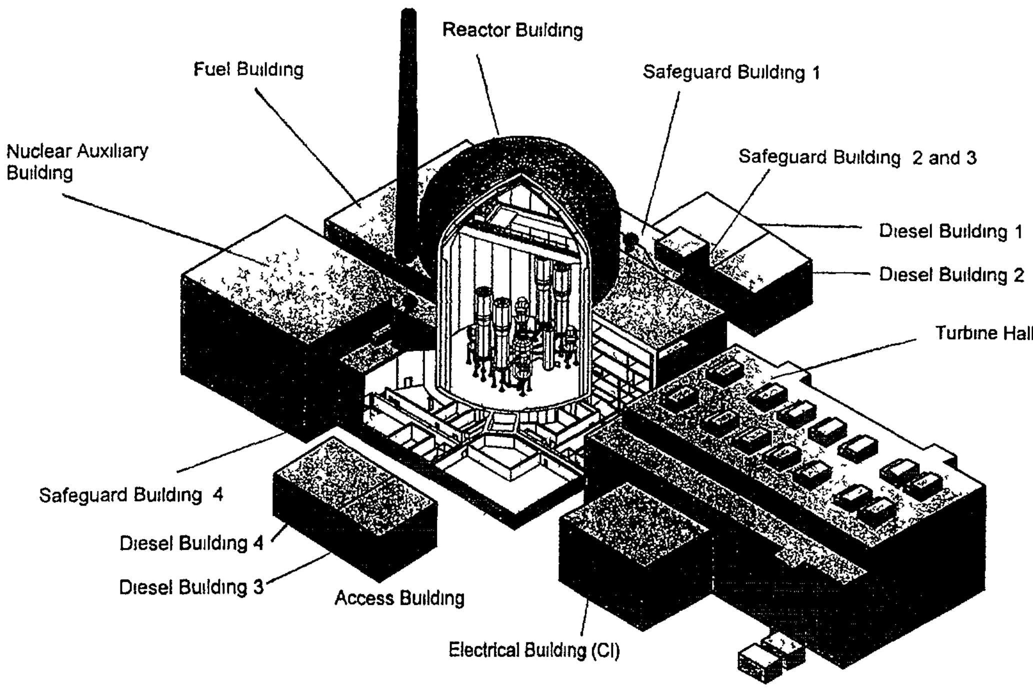

FIG. 4. EPR - Plot plan.

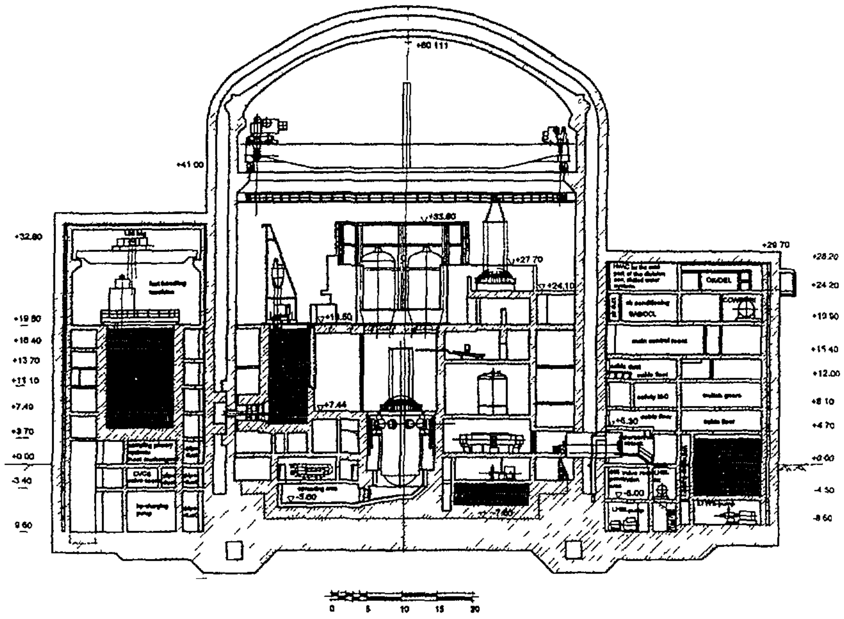

FIG 5 EPR - Buildıng arrangement, sectıon A-A.

To ensure containment leak-tightness, systems for isolation and retention and control of leakages are required. Leakages through the inner containment wall are released via the annulus air extraction system. Openings for personnel access or equipment supply to the inside of the containment are permanently closed hatches or air locks wıth double sealings on both sides.

This concept is also applied for penetrations of the HVAC systems. Fluid systems penetrating the containment are provided with double isolation valves, inside and outside the containment.

The structural integrity of the containment is protected by the thermal inertia of the concrete structures inside the containment (absorbing heat), and the safety injection system and the containment heat removal system (removıng heat).

Accumulation of combustible gases, especially hydrogen, is controlled. Furthermore, the basemat in the spreading compartment is protected by protective layers and the dedicated cooling system fed by the contaınment heat removal system against elevated temperatures resulting of a core melt.

The severe accident conditions described in the previous sections, lead to more severe design condıtions compared to the existing plants, and will thus result in an extrapolation of the design parameters, In this respect, the most important factor is the increased design pressure, which was defined as 650 kPa (6.5 bar abs). The prestressed concrete inner wall will also ensure capability to perform an integral leakage pressure test in air at design pressure.

4 3 1 Contaınment integrity

The requirement on limitation of the radiological consequences to the environment of the plant even under severe accident conditions implies strong demands on the containment as the last barrier for radioactive releases.

The maximum pressure and temperature reached during the most severe transient shall not exceed the design values, e.g., the design pressure of 650 kPa. The leak rate of the inner containment shall be lower than 1% of the containment volume per day under accident pressure.

It is important to control the formation of hydrogen, e.g., by zirconium-water reaction, and deflagration or detonation, the formation of non-condensable gases, e.g., as a consequence of corium-concrete interaction, and the residual heat generation by the molten core.

For residual heat removal from inside the containment after severe accidents the containment heat removal system is provided. Its prime function is to limit the pressure increase inside the containment below the design pressure, and to decrease this pressure to restrict the fission product releases through postulated containment leaks. For this purpose, a spray system with heat exchangers has been selected - with all active components located in a special compartment outside the containment.

The containment integrity and the core melt generated following severe accidents need to be controlled. Therefore the reactor pit design has been modified compared to earlier designs. The reactor pit bottom is by a slope connected to a spreading area which is provided to collect the core debris and separate it from the in-containment water storage tank to avoid steam explosion. In a later stage of the accident, water ingress is provided by dedicated melting plugs that allow water to cool the molten core material by passive means. The generated steam is condensed by the containment heat removal system exclusively provided for these accident sequences.

Interaction between concrete and the molten core material is prevented by a high temperature resistant protective layer on the reactor pit floor and the spreading area.

4.4 Other buildings

4.4.1 Safeguard and fuel buildings

Connections between the safety systems and the reactor coolant system are made as short as possible. The individual trains of the safety systems are arranged radially to the primary loops. Each train of the safety systems is protected against internal hazards from one train to another by location in specific separated areas, called divisions. The four trains of the safety systems are located in four safeguard buildings that surround the reactor building. The spent fuel pool is located in the fuel building which is separated from the safety divisions.

The reactor building, the fuel building and the four safeguard buildings are protected against external hazards, such as earthquake and explosion pressure wave. All these buildings are situated on a common raft.

Protection against aircraft crash is achieved by bunkerization of the safeguard buildings 2 and 3, the reactor building and the fuel building. The main control room and the remote shutdown station are also located in these bunkered safeguard buildings. The inner building structures of these buildings are decoupled from the outer shell in order to minimize induced vibrations.

The safeguard buildings 1 and 4 are not bunkered but geographically separated, so that only one division can be affected, the other remaining operable.

The storage pool for spent fuel assemblies is located outside the containment, facilitating fuel transport cask loading and unloading for the transport of fresh fuel to the plant and of spent fuel away from the plant. A transfer tube in the containment wall connects the inside of the containment with the fuel building.

The primary system is arranged symmetrically. Concrete walls are provided between the loops and between the hot and cold legs of each loop to provide protection against consequential failures. The pressurizer is located in a separate compartment. A concrete wall around the entire primary system protects the containment from missiles and reduces the radiation from the primary system to the surroundings.

A water pool for storage of the upper core internals during refuelling, and for the entire core internals during inspection, is provided inside the containment for radiation protection reasons.

4.4.2 Nuclear auxiliary building

This building houses mainly

- Boron recycle system (coolant and demineralized water storage, coolant treatment and coolant purification)

- Fuel pool purification system
- Gaseous waste system
- Steam generator blowdown system (including purification)

4.4.3 Diesel buildings

The four diesel buildings constitute the four redundancies of the emergency power supply system. Diesel buildings 1/2 and 3/4 are located at opposite positions adjacent to the reactor building and are protected against external hazards by physical separation.

The diesel buildings 1 and 4 are housing also the smaller diesel generator sets.

5. TECHNICAL DATA

General plant data		
Power plant output, net	~1750	MWe
Reactor thermal output	4900	MWt
Power plant efficiency, net	36	%
Nuclear steam supply system		
Number of coolant loops	4	
Primary circuit volume, including pressuriser	380+75	m^3
Steam flow rate at nominal conditions	2776	kg/s
Steam temperature/pressure	289/7.36	°C/MPa
Feedwater temperature/pressure	230/7.36	°C/MPa
Reactor coolant system		
Primary coolant flow rate	22240	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	292.5	°C
Coolant outlet temperature, at RPV outlet	330	°C
Reactor core		
Active core height	4.2	m
Equivalent core diameter	3.8	m
Heat transfer surface in the core	7975	m^2
Fuel inventory	124	t U
Average linear heat rate	17.86	kW/m
Average core power density (volumetric)	103	kW/l
Thermal heat flux, F_q	598	kW/m^2
Fuel material	Sintered UO_2	
Fuel assembly total length	4 800	mm
Rod array	square, 17x17	
Number of fuel assemblies	241	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	25	
Number of spacers	9	
Enrichment (range) of first core	≤ 5.0	Wt%
Operating cycle length (fuel cycle length)	12-18	months
Average discharge burnup of fuel	65000	MWd/t
Cladding tube material	Zr 4	
Cladding tube wall thickness	0.625	mm
Outer diameter of fuel rods	9.5	mm
Active length of fuel rods	4200	mm
Burnable absorber, strategy/material	Gadolinium	
Number of control rods	89	
Absorber rods per control assembly	24	
Absorber material	Ag-80, In-15, Cd-5	
Drive mechanism	Magnetic jack	
Positioning rate [10x75 or 750 mm/min]	75	steps/min
Soluble neutron absorber	boron	

Reactor pressure vessel

Cylindrical shell inner diameter		4870	mm
Wall thickness of cylindrical shell		250 + 7.5	mm
Total height		13105	mm
Base material:	cylindrical shell	16MND5/20MnMoNi55	
	RPV head	"	
	cladding	stainless steel	
Design pressure/temperature		17.6/351	MPa/°C
Transport weight (lower part)		405	t
	RPV head	115.5	t

Steam generators

Type	U-tube heat exchanger	
Number	4	
Heat transfer surface	8171	m^2
Number of heat exchanger tubes	5980	
Tube dimensions	19	mm
Maximum outer diameter	20.1	mm
Shell and tube sheet material	16MND5/20MnMoNi55	
Tube material	Incoloy 800 or Inconel 690	

Primary containment

Type	prestressed concrete	
Overall form (cyl.)	cylindrical	
Dimensions (diameter/height)	53/45	m
Free volume	80000	m^3
Design pressure/temperature (DBEs)	~ 650/170	kPa/°C
(severe accident situations)	~ 650/170	kPa/°C
Design leakage rate	< 1	vol%/day
Is secondary containment provided?	yes, reinforced concrete (APC-Protection)	

6. PROJECT STATUS

At the end of the century, NPI will be ready for marketing of the EPR and construction of the first plant may start in the beginning of year 2000. No specific site has yet been selected.

IAEA-SM-353/42

ADVANCED BOILING WATER REACTOR

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Abstract

In the Boiling Water Reactor (BWR) system, steam generated within the nuclear boiler is sent directly to the main turbine This direct cycle steam delivery system enables the BWR to have a compact power generation building design Another feature of the BWR is the inherent safety that results from the negative reactivity coefficient of the steam void in the core. Based on the significant construction and operation experience accumulated on the BWR throughout the world, the ABWR was developed to further improve the BWR characteristics and to achieve higher performance goals. The ABWR adopted "First of a Kind" type technologies to achieve the desired performance improvements. The Reactor Internal Pump (RIP), Fine Motion Control Rod Drive (FMCRD), Reinforced Concrete Containment Vessel (RCCV), three full divisions of Emergency Core Cooling System (ECCS), integrated digital Instrumentation and Control (I&C), and a high thermal efficiency main steam turbine system were developed and introduced into the ABWR.

1. INTRODUCTION

The first ABWR plants were constructed for the Tokyo Electric Power Company (TEPCO) Kashiwazaki-Kariwa Nuclear Power Station Units No.6 and 7 (K-6 and K-7) as a joint venture effort of GE, Hitachi, and Toshiba. The construction started in September 1991 and February 1992, with commercial operation starting in November 1996 and July 1997, respectively. Both units have now completed the first full fuel cycle of operation, and the desired performance improvements have been achieved by these first units. A view of the site area is shown in figure 1.

The following provides a general description of the ABWR and the construction experience of the units at K-6 and K-7.

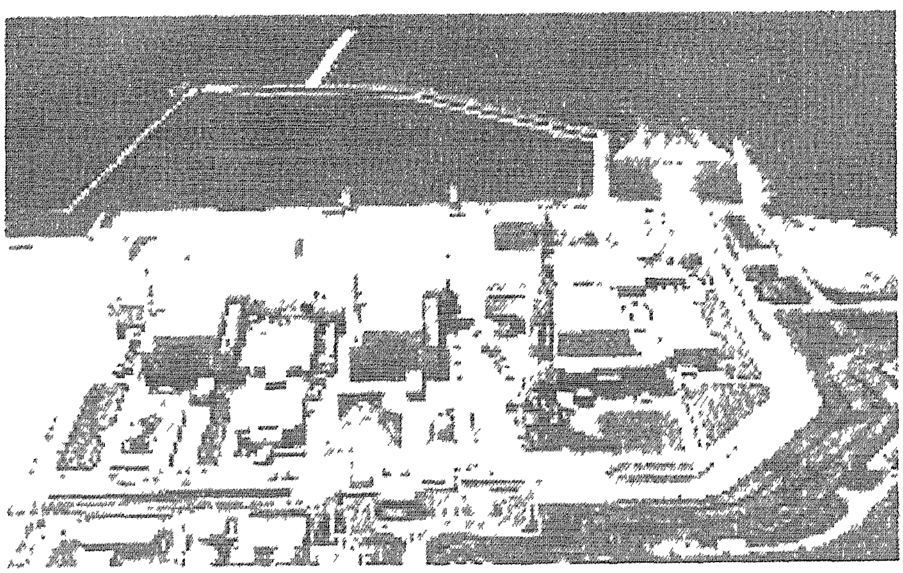

FIG. 1. View of TEPCO Kashiwazaki-Kariwa N.P.S. Unit 5, 6, 7 (from right to left).

2. GENERAL SPECIFICATION

The ABWR was designed to have the range of 1,350 - 1,380 MW electrical output with a nuclear boiler thermal output of 3,926 MW. The actual K-6 and K-7 gross electrical output is 1,356 MW based upon the local sea water temperature. The thermal output is generated by the nuclear core that consists of 872 fuel bundles.

The ABWR was designed as a twin unit plant. Each unit has its own Reactor Building and Turbine Building while the Control Building, Service Building, Radioactive Waste Building, and other utility facilities are shared by the two units. The Reactor Buildings contain the RCCV, nuclear boiler systems, and the ECCS systems. The Turbine-Generator, main condenser, and associated condensate and feedwater systems are located in the Turbine Building. In case of K-6 and K-7, the Circulating Water Pump and auxiliary sea water cooling systems are located in the heat exchanger area which is a part of the Turbine Building. The Main Control Panels for both K-6 and K-7 are located in the Main Control Room of the Control Building so that one operating crew can operate both units. The Radioactive treatment systems and the Condensate Storage Pool are located in the Radioactive Waste Building. The Service Building is used for the protective clothing changing area and check points. The two Turbine Buildings and the Radioactive Waste Building are connected together by a single crane rail to enable all these buildings to be available for the turbine generator lay-down space during the refueling outages.

The ABWR design objectives included a 40 year plant life, 87% or better plant availability, less than 0.36 man-Sv per year occupational radiation exposure, less than 10^{-7} /reactor year core damage frequency.

The ABWR general specification is shown in Table 1, and the overall system diagram is shown in Figure 2.

TABLE-1 ABWR Major Plant Specification

ITEM	ABWR	Conventional BWR
Plant Output		
Thermal Output	3 926 MW	3.293 MW
Electric Output	1 356 MW	1,100 MW
Vessel Dome Pressure	7 17 MPa	7 04 MPa
Main Steam Flow	7.640 t/h	6.410 t/h
Turbine	TC6F-52in	TC6F-41in
Reheat	Two-stage Reheat	No Reheat
Power Density	50 6 kW/l	50 0 kW/l
Fuel Assembly	872	764
Control Rods	180-B_4C+25-Hf	168-B_4C+17Hf
Control Rod Drive		
Normal Operation	Fine Motion Stepping Motor	Hydraulic Locking Piston
Scram Insertion	Hydraulic with Stepping Motor Backup	Hydraulic
Recirculation System	10 Internal Pumps	2 External Pumps with 20 Jet Pumps
Core Flow (100%Rated)	52 200 t/h	48.300 t/h
Emergency Core Cooling System	3 Divisions	3 Divisions
Division 1	RCIC+LPFL	LPCS+LPCI
Division 2	HPCF+LPFL	LPCI+LPCI
Division 3	HPCF+LPFL	HPCS
Primary Containment		
Type	Pressure Suppression	Pressure Suppression
Configuration	Cylindrical Reinforced Concrete Containment Vessel	Conical Freestanding Steel Vessel
Control and Instrumentation	Microprocessor based digital circuit	Relay-analog circuit
Data Transmission	Intelligent multiplexed fiber optics	Hard-wired cable

RCIC Reactor Core Isolation Cooling HPCF High Pressure Core Flooder
LPCS Low Pressure Core Spray LPCI Low Pressure Coolant Injection
LPFL Low Pressure Flooder

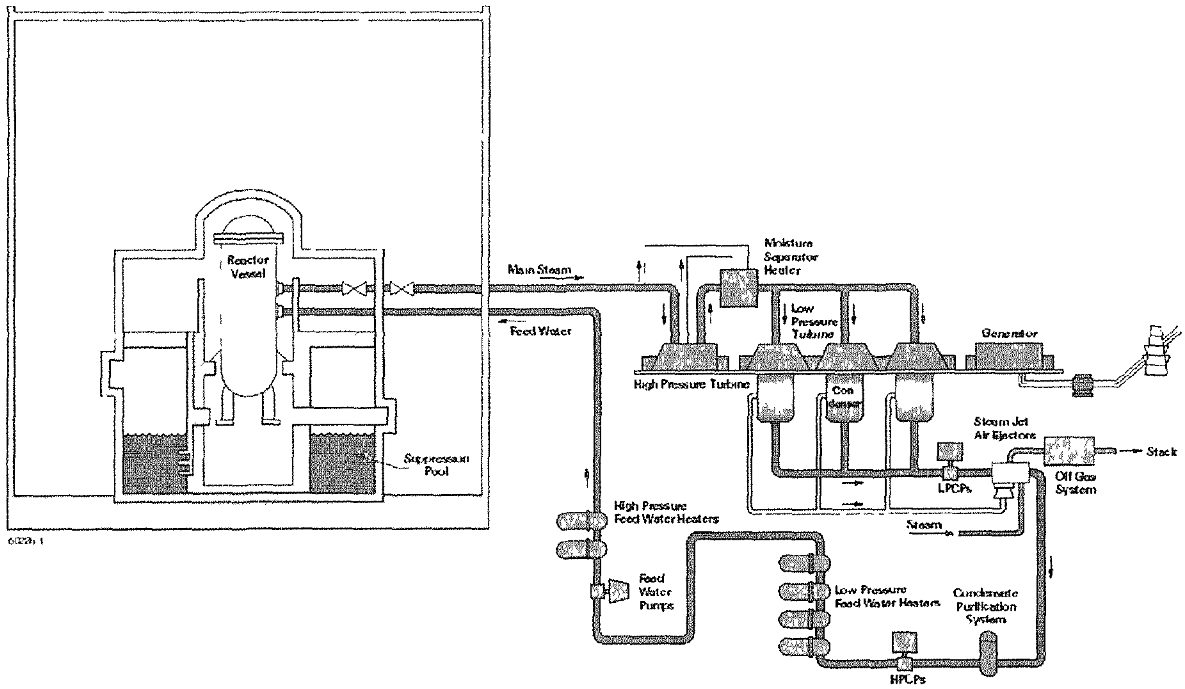

FIG. 2 ABWR overall system diagram

3. SAFETY FEATURES

3.1 Core Damage Frequency

In postulated Loss of Coolant Accident (LOCA) scenarios, core uncovery may result in core damage. Conventional BWRs utilize a defense-in-depth approach and their evaluated core damage frequency is less than 10^{-6}/ reactor year, which is already less than the 10^{-5}/ reactor year that is presented in the IAEA guidelines. However, for the ABWR, an even lower core damage frequency was desired.

A break in the piping which is connected to the Reactor Pressure Vessel (RPV) can result in a LOCA event. In case of a conventional BWR, the Primary Loop Recirculation (PLR) pump suction line is the largest pipe connected to the RPV and the postulated break of that piping is the limiting event for design basis LOCA analysis.

The ABWR adopted the Reactor Internal Pump (RIP) for the primary recirculation system instead of the external PLR design. With the elimination of the PLR lines, there are no longer any large bore pipes that connect to the RPV at an elevation below the top of the reactor core. Postulated LOCA scenarios were analyzed and the result was that a break in High Pressure Core Flooder (HPCF) line was determined to be the most severe event for the ABWR. Even in this case of an HPCF line break, it has been shown through analysis that the ECCS operation is sufficient to maintain the reactor coolant level above the top of the core. The Core Damage Frequency of the ABWR is calculated to be less than 10^{-7}/ reactor year.

With the elimination of PLR large bore piping, reactor pressure remains relatively high during the LOCA event. To assure early ECCS core injection, enforcement of the high pressure injection system was desired. The ECCS network in the ABWR is composed of three high pressure injection systems and three low pressure injection systems. However, the capacity of each system has been reduced from that in a conventional BWR as a result of the design basis LOCA flow rates being significantly reduced in the ABWR design.

3.2 Anticipated Transient Without Scram

The rapid Control Rod (CR) insertion into the core (scram) is initiated to terminate or mitigate a broad range of plant transient and postulated accident conditions. The Control Rod Drive (CRD) system has been designed to achieve that scram function with a high degree of assurance. Nevertheless, the Anticipated Transient Without Scram (ATWS), in which the scram function is postulated to fail to function properly, is considered in the ABWR design.

In a conventional BWR, hydraulic pressure is used for the normal insertion and withdrawal of the CR, and also for the scram function. Under normal scram conditions, high pressure water that is stored in an accumulator is discharged to the underside of the Control Rod Drive (CRD) causing it to rapidly insert (i.e., scram). The discharge of that high pressure water to each CRD is controlled by a Scram Valve. The Scram Valve is an air-operated valve that is actuated by the operation of a Scram Solenoid Valve (SSV). The Reactor Protection System (RPS) controls the operation of the SSV. In an ATWS event, the SSVs are postulated to have failed such that they do not operate to release the air pressure holding the Scram valves shut. As an ATWS mitigating measure, backup scram solenoid valves (of a diverse design) are installed on the Instrument Air line. Operation of the backup scram solenoid valves by the RPS causes the entire air supply line to be depressurized so that pressure controlling the individual Scram Valves is released and the high pressure water can be delivered to the CRDs as required for the scram operation.

The ABWR adopted the FMCRD which uses a stepping motor for the normal CR insertion and withdrawal function. But the hydraulic pressure is used for the scram function like a conventional CRD system. The back up scram valves, like those in a conventional BWR, are also applied. In addition to that back-up scram feature, in the ABWR, the stepping motor is activated automatically upon scram initiation. This motor driven CR run-in function provides an additional degree of ATWS mitigation.

3.3 Severe Accident Mitigation

Features which specifically address Severe Accident mitigation have been incorporated into the ABWR design. To provide an additional means of injecting water to the core under severe accident scenarios, facilities are provided to inject Make Up Water (MUW) system water to the RPV via connections in the Residual Heat Removal (RHR) system piping. To provide additional means for replenishing the water available to the ECCS special facilities are provided to connect the Fire Protection (FP) system to the MUW system and to connect the MUW system to the Suppression Pool which is inside the RCCV and is the primary suction source for the ECCS. Finally, to avoid over-pressurization of the RCCV, a feature has been added to the vent system to bypass the Standby Gas Treatment System (SGTS) filter trains so that the RCCV can be vented directly through the main stack.

4. CORE AND FUEL

The latest Japanese BWR fuel bundle design consists of an 8 x 8 or 9 x 9 array of fuel pins surrounded by a fuel channel box. Inside each fuel pin, small pellets of sintered Uranium oxide are stacked. The fuel pins are supported by upper and lower tie plates at both ends and an intermediate positions by spacer assemblies to form a fuel bundle. The fuel channel box forms a flow channel for the reactor coolant. Throughout the development of fuel performance improvements, BWR fuel has maintained its basic physical dimensions so that application of new fuel designs to the existing fleet of operating BWRs is always possible.

In previous fuel designs, the uranium concentration was uniform alongside axial location of the fuel pin. Maintaining power as the fuel burns up is accomplished by the relatively frequent insertion and withdrawal of Control Rod throughout the core. However, the current advanced fuel designs have variable Uranium concentrations axially from the upper regions of the individual fuel pins to the lower regions. With this new type of fuel and the application of Control Cell core management methods, only the Control Rods in the Control Cell locations are used for power compensation. The ABWR has 25 Control Cells, and Control Rods with Hafnium blades are used in these Control Cell locations. Conventional Control Rods with Boron Carbide blades are used in the other Control Rod locations, which are used only for the purpose of plant shut down.

The BWR core is unique, in comparison to other Light Water Reactor (LWR) designs, in that the steam voids generated within the core perform an important role in controlling the core reactivity. The uranium fission process generates only fast neutrons and, in order to achieve a sustained chain reaction, these fast neutrons must be moderated to become thermal neutrons. In the BWR core, water acts as the moderator. When the steam void increases in the core, reactivity will decrease because of the decreased moderation function of the water. Increased RIP speed, which corresponds to increased core flow, will drive steam voids out from the core. In this condition, the reactivity will increase due to the increased water volume. Reactor power change is accomplished by capitalizing upon this inherent BWR negative power coefficient. An increase in recirculation flow temporarily reduces the volume of steam in the core by removing the steam voids at a faster rate. This increases the reactivity of the core, which causes the reactor power to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and a new constant power level is established. When the recirculation flow is reduced, the power level is reduced in a similar manner.

5. POWER GENERATION

5.1 Nuclear Boiler

Moisture entrained in the steam, which is generated within the core, must be removed before the steam is delivered to the turbine. To remove that moisture, a Steam Separator and a Steam Dryer are located inside the RPV directly above the core. The steam exiting the core flows upward into the Steam Separator, in which the moisture is removed using the upward centrifugal movement of the steam. The Steam Dryer, which is located directly above the Steam Separator, further removes the moisture using the impingement vanes. The dry steam exits the RPV through a flow limiting orifice in the RPV nozzle of the Main Steam Line. The moisture accumulated within the Steam Separator and Steam Dryer is discharged to the annulus region between the Core Shroud and RPV, and returned to the core by the Reactor Internal Pump.

6. TURBINE GENERATOR

The ABWR has a 1,540MVA capacity Main Generator which is driven by three Low Pressure Turbines and one High Pressure Turbine in tandem compound configuration.

The steam from the nuclear boiler goes into the High Pressure Turbine through the Main Stop Valve and Control Valve. The High Pressure Turbine exhaust steam passes through the Moisture Separator ReHeater (MSH). The MSH removes the moisture from the steam, and the dry steam is heated by two stages of extraction steam. Within the MSH, the moisture is removed utilizing impingement vanes similar to the Steam Dryer. The first stage heating steam is extracted from the High Pressure Turbine and the second stage is extracted from the main steam piping upstream of the High Pressure Turbine. The reheated steam is discharged to the Low Pressure Turbine through the Combined Intermediate Valve.

The Low Pressure Turbine adopted in the ABWR is one of the largest of its type. The conventional 1,100 MW class BWR have Low Pressure Turbines that have last stage blades that are 41 or 43 inches long, depending on the grid frequency. To achieve improved thermal efficiency, the ABWR adopted Low Pressure Turbines that have last stage blades that are 52 inches long. The Low Pressure Turbine exhaust steam goes downward into the Main Condenser where the steam is cooled by the sea water.

Three Circulating Water pumps send the sea water to heat removal units within the plant. In the Main Condenser, the sea water goes through titanium tubes to condensate the turbine exhaust steam. A Ball Cleaning facility is utilized to maintain the cleanliness of these titanium tubes.

6.1 Condensate and Feedwater System

The Condensed water is returned to the RPV by first passing through three 50% capacity Low Pressure Condensate Pumps, with one pump normally in stand-by. The condensate then passes through the Steam Jet Air Ejectors (SJAE). The SJAEs are used to remove non-condensable gasses from the condensate water and it is driven by auxiliary steam or the house boiler steam. The de-aerated water passes through the Gland Steam Condenser, which condenses the Turbine gland steam, and then it is filtered and demineralized by the Condensate Filter (CF) and the Condensate Demineralizer (CD).

The CF and CD remove foreign particles and ions from the condensate water to maintain the water quality. In the BWR system, it is very important to maintain the water quality from the view point of the corrosion prevention and occupational exposure reduction since the condensate water passes directly through the core. Three hollow fiber filter CFs remove particles from the water. The pre-coat type CF that is commonly used in a conventional type BWR has the capacity to remove both particles and ions. However, the pre-coat material contributed significantly to the generation of radio-active waste and so, the non pre-coat type filter was developed in Japan. The hollow fiber filters have been used at several of the latest Japanese BWRs constructed and their operating experience has been excellent. The resin bed Condensate Demineralizer performs the final removal of particles and ions from the water.

The demineralized water is pumped by the High Pressure Condensate Pumps through the four stages of Low Pressure Feedwater Heaters, which are located in the upper portion of the Main Condenser. Each of the three High Pressure Condensate Pumps have 50% capacity, with one pump normally in stand by. Each stage of the Low Pressure Feedwater Heaters has three heaters which corresponds to the three condenser rooms. The water exiting the Low Pressure Feedwater Heaters is pumped by the Reactor Feedwater Pump through the High Pressure Feed Water Heaters. Two 50% capacity Turbine Driven Feedwater Pumps are used during normal plant operations while other, i.e., 25% capacity motor driven feedwater pumps, are used during plant start-up and transient conditions. Two-stage High Pressure Feedwater Heaters are used for the final step in the heating of the feedwater.

In the conventional Japanese BWR plants, all the feedwater heater drains are discharged back to the Main Condenser. For ABWR, feedwater heater drain pump systems were adopted. The High Pressure Feedwater Heater drains are discharged into High Pressure Drain Pump system. The accumulated water in the high pressure drain tank is pumped by the High Pressure Heater Drain Pump into the condensate water system upstream of the feedwater pumps. The Low Pressure Feedwater Heater drains are cascaded into the Low Pressure Drain Pump system. The accumulated water in the low pressure drain tank is pumped by the Low Pressure Heater Drain Pump into the condensate water system upstream of the CD. With the application of these heater drain pump systems, overall system thermal efficiency is improved and the condensate water system capacity is reduced.

6.2 Electrical System

The 1,540MVA capacity Main Generator is driven by the Turbine. The generator consists of a stator (stationary armature winding and core) and rotating field. The excitation system provides DC current to make a rotating field within the generator. The Automatic Voltage Regulator (AVR) regulates the output Voltage of the generator at 27 kV.

The generator output voltage is then converted by the Main Transformer to 500 KV ultra-high voltage for connection to the electrical power grid.

During power operation, the station load is supplied from the Main Generator via two House Transformers. Safety-related equipment is connected to the off-site power network via a Start-up Transformer, and to the Emergency Diesel Generators.

7. EMERGENCY SYSTEM

7.1 Emergency Core Cooling System (ECCS)

The BWR ECCS system has had many variations. The latest BWR-5 units have three divisions of ECCS configured as shown below. The BWR-5 includes a Reactor Core Isolation Cooling (RCIC) system, but this system is not classified as an ECCS. Each division has its own Diesel Generator for emergency power supply.

- Division 1: Low Pressure Core Spray (LPCS) and Low Pressure Core Injection (LPCI)
- Division 2: LPCI and LPCI
- Division 3: High Pressure Core Spray (HPCS)
- Automatic Depressurization System (ADS)

The ADS is one of the Safety Relief Valve (SRV) functions and its purpose is to rapidly depressurize the reactor pressure so that Low Pressure Core Injection (LPCI) system operation is possible. The LPCI system is also used as a Residual Heat Removal (RHR) system during the normal plant refueling outages.

The ABWR strengthened the high pressure core cooling capability and has three full divisions of ECCS which are configured as shown below. The RCIC has been upgraded to be an ECCS. Through development study, it was confirmed that core spray is not mandatory to assure uniform core cooling. Based upon those results, all high and low pressure injection system employ core-flooders, instead of core sprays.

- Division 1: RCIC and Low Pressure Flooder (LPFL)
- Division 2: High Pressure Core Flooder (HPCF) and LPFL
- Division 3: HPCF and LPFL
- ADS

Each division has its own Diesel Generator for emergency power supply.

The HPCF directly injects water inside of the Core Shroud, while the RCIC provides water to the RPV through the feed water lines. Both of the high pressure injection systems uses the Condensate Storage Pool as their primary water source and the Suppression Pool as their secondary water source.

The LPFL has four different function modes. The low pressure core injection mode is used for the ECCS purpose. Other major operation modes are (1) shut down cooling mode, which is used during the plant shut down; (2) containment spray mode, which is used to suppress steam released into the Drywell during postulated LOCA events, and; (3) suppression pool cooling mode, which is used to cool the Suppression Pool water.

7.2 Reactor Containment

The free-standing steel Primary Containment Vessel (PCV) has been used in previous BWR designs. The PCV is designed to hold any fluids that could be discharged from the primary coolant system during a postulated LOCA event. The BWR uses a pressure suppression type containment, and a large volume of water is maintained in the PCV as a Suppression Pool.

In a conventional BWR-5, the PCV is separated by a diaphragm into a Drywell volume and a Wetwell volume. The Wetwell contains the Suppression Pool. Located in the Drywell is the RPV; the main steam and feedwater piping and; related valves, such as Main Steam Isolation Valve (MSIV) and SRV.

In case of a LOCA event, the steam released would fill the Drywell and would be discharged to the Suppression Pool through a series of connecting vent pipes. The released steam is condensed in the Suppression Pool. With this pressure suppression function, the PCV pressure is kept relatively low even during a LOCA event.

A drawback associated with the use of this steel PCV is that Reactor Building construction could not proceed until the PCV leak test is complete. Also the configuration restriction arising from the steel structure presented plant and equipment arrangement restrictions. To eliminate such issues, the Reinforced Concrete Containment Vessel (RCCV) was adopted for the ABWR. Instead of a 38 mm thick steel structure, as in the case of the conventional PCV, in the RCCV design a 6.4 mm steel liner plate is used for securing leak-tightness and 2 meter of reinforced concrete behind the liner plate is used for pressure containment. The reinforced concrete of the RCCV is integrally interconnected with the concrete floors and walls of the Reactor Building. With this configuration, the building construction can proceed simultaneously with the RCCV construction and there is greater flexibility in the design of the Drywell arrangement.

The conventional BWR has two big Primary Loop Recirculation (PLR) pumps and external loop piping which connect the pumps to the RPV. In the ABWR design, these external pumps and associated piping are replaced by the Reactor Internal Pumps (RIP) which are directly attached to the RPV. These changes in the reactor recirculation system enabled the required volume of the primary containment to be reduced and, at the same time, the RCCV center of to be lowered.

The Flammable Gas Control (FCS) system consists of a blower, heater, recombiner, and cooler. The FCS system is installed to prevent an explosive reaction between the hydrogen and oxygen that might be generated during a postulated LOCA event. Through operation of the FCS, the flammable gases are re-combined and the water produced is returned to the Suppression Pool.

The Standby Gas Treatment System (SGTS) consists of two trains of equipment. Each train consists of a dehumidifier, a high efficiency particulate filter, Iodine charcoal filter, and exhaust fan. The SGTS is installed to maintain the Reactor Building at a negative relative pressure and to remove suspended radioactive materials in the Reactor Building.

8. INSTRUMENTATION AND CONTROLS

The overall control system configuration diagram for the ABWR is shown in Figure 3.

8.1 Reactor Control Systems

The Reactor Control Systems are composed of three major systems. The Reactor Power Control System controls the reactivity, the Reactor Pressure Control System controls the reactor pressure, and the Reactor Feedwater Control System controls the reactor water level.

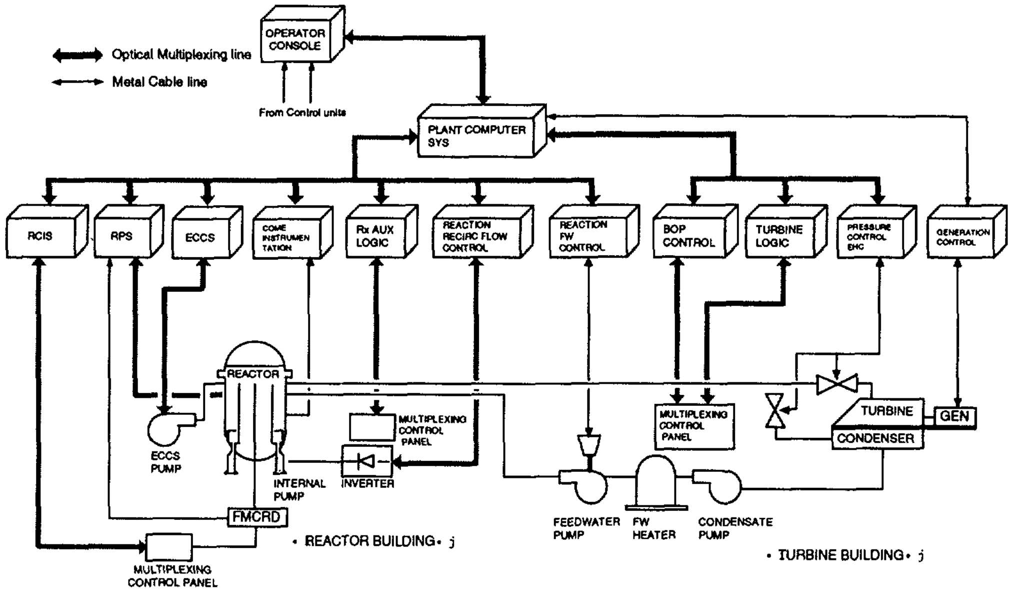

FIG. 3 ABWR Control System Configuration Diagram

8.1.1 Reactor Power Control System

The Reactor power is controlled by three elements: Control Rods position, recirculation flow, and the master controller signal from the main turbine.

The Rod Control and Information System (RC&IS) sends signals to the individual Fine Motion Control Rod Drive (FMCRD) mechanisms to withdraw or to insert. All Control Rods are inserted into the core when the plant is in shut down conditions. To start the plant operation, Control Rods withdrawal is necessary to increase the core reactivity by reducing the neutron absorption. With the application of the FMCRD and the associated RC&IS, multiple Control Rods can be simultaneously withdrawn (and inserted) and, as a result, less time is required to transition the plant from shutdown to normal operating conditions.

The core flow rate depends on the RIP rotating speed. The RIP is driven by an individual inverter controlled electric power source called an Adjustable Speed Drive (ASD). The Recirculation Flow Control (RFC) system sends a signal to control the RIP rotating speed based on the RFC master controller signal. Approximately 35% of the rated power level can be adjusted by the RFC control without any CR movement.

A signal from the speed-load governing mechanism to the master flow controller establishes the necessary reactor recirculation flow required to meet the system power requirements.

8.1.2 Reactor Pressure Control

The reactor pressure is controlled by the turbine Control Valve (CV) and Turbine Bypass Valve (TBV) opening control using the Electro Hydraulic Controller (EHC).

8.1.3 Reactor Feedwater Control System

The Feedwater Control System (FWCS) automatically controls the feedwater flow into the RPV to maintain the water at a predetermined level during all modes of plant operation. The FWCS utilizes signals from reactor vessel water level, steam flow, and feedwater flow.

8.2 Reactor Protection System

The Reactor Protection System (RPS) initiates CRD scram to insert all the Control Rods in the event of an abnormal plant condition.

The RPS is a four channel system which employs "two out of four" logic. This system is designed for maximum reliability to assure scram initiation is performed when necessary while minimizing the potential for unnecessary scrams.

Typical scram signals are listed below.

- High Pressure in the PCV.
- Low Water Level in the RPV.
- High Reactor Pressure.
- High Neutron Flux.
- Fast Closure of Turbine Control Valves.
- Closure of Turbine Main Stop Valve.
- Main Steam Isolation Valve Closure
- Steam Line High Radio-activity
- Manual Scram

8.3 Neutron Monitoring System

The reactor power is monitored by a coordinated set of neutron monitoring (sub)systems which come into service depending on the neutron flux level.

8.3.1 Source Range and Intermediate Range

At plant start up, when the neutron flux level is quite low, neutron flux is monitored by the Source Range Neutron Monitor (SRNM) using a neutron countrate method. When the core reaches the intermediate neutron level, the countrate method is automatically changed to the voltage variance method.

The ABWR has 12 SRNM monitors, and they are inserted into the middle of the core through inverted thimbles.

8.3.2 Local Power Range Monitor

During power operation, the neutron flux is monitored by the Local Power Range Monitor (LPRM). There are 52 LPRM assemblies located throughout the core. Each LPRM assembly has four fission chamber detectors, located at fixed positions vertically in the core, and a calibration guide tube for a Traversing In-core Probe (TIP).

8.3.3 Average Power Range Monitor

The Average Power Range Monitor (APRM) is a four channel system which monitors the bulk average power of the core. Within each channel of APRM, the neutron flux signals from 52 LPRM detectors are averaged and the bulk power of the core determined.

8.3.4 *Traversing In-Core Probe*

For the calibration of the LPRM, the Traversing In-core Probe (TIP) ion chamber is inserted into each calibration tube of the LPRM assembly. Traversing the ion chamber through the LPRM develops a vertical neutron flux profile upon which the LPRM detectors at that location can be calibrated. The TIP index mechanism selects the LPRM to which the TIP is inserted.

8.4 The State-of-the-Art I&C Technology

8.4.1 *Integrated Digital Control Technology*

Japanese BWRs have made steady progress on the application of digital technology. It started with application to the radioactive waste treatment system controls, and then was applied to other non-safety systems. In the ABWR, safety related systems also apply digital technology and, thus, achieving a totally digital plant for the first time. The objectives for digital system application were;

- higher reliability with less components, self-diagnosis features, and elimination of sensor drift, and
- better maintainability with modular replacement, and
- reduced construction cost through reduced number of control panels and use of fiber optic multiplexing data system to reduce the amount of cabling required.

8.4.2 *ABWR Main Control Room Panels*

The ABWR main control panel has two major features. One of them is the operator console that has Cathode Ray Tubes (CRT) and Flat Displays (FD). Control of the entire plant can be completed from this operator console. The other feature is the Large Display Panel (LDP). The LDP is a solid mimic with the status of all major plant systems and equipment shown in the center of the panel. Important plant level annunciators are displayed at the left side of the panel while system annunciators are displayed on the top of the panel. The 110 inch large screen display is located on the right side of the panel on which any information from the individual CRTs can be displayed.

Plant operation is conducted by touch operation on the CRT or FD screens. The operator can access the individual control screens through multi-layer screen operation.

Operator guidance and automatic operation functions have been widely applied throughout the ABWR control room design. The operator work load was analyzed during the ABWR development stage. Through these studies, activities of heavy work load (such as plant stabilizing control after a scram) were identified and such heavy work load activities were automated.

During the plant start up, CR withdrawal and associated operations are conducted automatically after operator initiation. There are a series of pre-determined hold points where the plant condition will be monitored. At these hold points, the automated operation will stop and wait for the operator confirmation and action before automated actions can continue.

9. PLANT CONSTRUCTION AND OPERATION

The first ABWR was adopted by Tokyo Electric Power Company's Kashiwazaki-Kariwa Nuclear Power Station Units Nos. 6 and 7. The construction was completed as a joint venture effort of GE, Hitachi, and Toshiba.

The construction was achieved through the use of the latest construction technology, such as the all-weather construction method and the large block module construction method. During the plant pre-operational and start-up testing, it was confirmed that all the pre-determined ABWR performance goals were achieved.

K-6 started commercial operation on November 7, 1996 and K-7 on July 2, 1997. The first fuel cycle of operation for both units have been completed and the operating performance was excellent.

10. CONCLUSION

The first ABWR plants completed construction and started commercial operation in Japan. The construction experience and the operating performance showed the expected performance goals are achieved. With this result, two more subsequent ABWR units are now under licensing process and several ABWR units are in the planning stage in Japan.

Beyond Japan, the U.S. N.R.C. has issued the Design Certificate for the ABWR, two ABWR units have started construction in Taiwan, China, and the major components for these units are being manufactured.

It is expected that the ABWR will be the major nuclear power plant design in the coming century.

XA0053574

SYSTEM 80+™ STANDARD PLANT — DESIGN AND OPERATIONS OVERVIEW

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Abstract

The System 80+ Standard Plant Design is a 1400 MWe evolutionary Advanced Light Water Reactor (ALWR), designed to meet the Electric Power Research Institute (EPRI) ALWR Utility Requirements Document (URD) and the demands of the international market for nuclear power plants which are not only safer but also more economical to maintain and operate. ABB Combustion Engineering Nuclear Power used a defense-in-depth process that (1) adds design margin to basic components to improve performance during normal operation and to decrease the likelihood of an unanticiapted transient or an accident, (2) improves the redundancy and diversity of safety systems in order to mitigate design basis accidents and prevent severe accidents, and (3) improves severe accident mitigation capability. This paper describes the most important improved systems and components with emphasis on severe accident prevention and mitigation capability. The improved design features were implemented in an evolutionary manner using proven components. This approach ensures that the plant operates safely and economically, as demonstrated by operating plants in the US and the Republic of Korea. Detailed studies, summarized in this paper, have shown that the System 80+ plant availability is expected to exceed the ALWR requirement of 87% and that the annual operations and maintenance costs are expected to be reduced by $14 million.

1. INTRODUCTION

The System 80+ Standard Plant Design was developed by incorporating improvements into the System 80® design already in operation at the Palo Verde Nuclear Generating Station in Arizona, USA. To improve safety, the design incorporates a balanced measure of design margin, accident prevention, and accident mitigation. The design complies with the EPRI goals of simplicity, improved reliability, improved accident prevention and mitigation, improved economics, and better man-machine interfaces. Unproven features that could slow the licensing process and/or construction or could result in unreliable operation have been avoided. Accordingly, the design does not require any prototype testing. Human Factors Engineering was extensively used to enhance the layout and performance capabilities of the completely integrated, computer-based Nuplex 80+™ Advanced Control Complex.

As required by US Nuclear Regulatory Commission (NRC) regulation, 10 CFR 52, the scope of the System 80+ design covers an essentially complete nuclear power plant and includes all structures, systems, and components that can significantly affect safe operation. The design also contains the level of detail necessary to complete NRC review and the preparation of procurement, construction, and installation specifications. Certification of the System 80+ design by the NRC was awarded in May 1997. The 10 CFR 52 approach provides a process for resolving licensing issues related to the design before any commitment to construction. A utility can reference a Certified Design and apply for a single combined license, authorizing both construction and operation, with assurance that the NRC will accept the certified portion of the design without further review. Furthermore, any public hearings undertaken on a specific combined operating license application would exclude issues related to the certified portion of the standard plant design. This process will allow owner/operators to plan for new nuclear plants by reducing the uncertainty associated with regulatory delays or design modifications during plant construction and startup. An overview of the System 80+ design is shown in Figure 1.

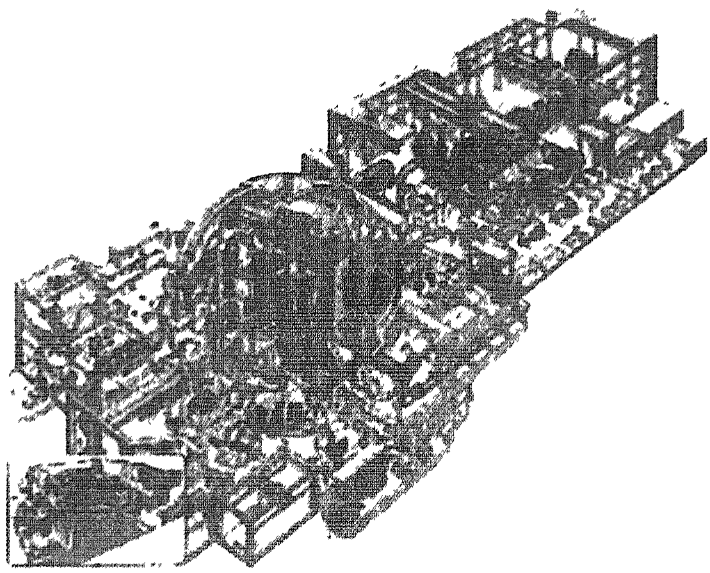

Figure 1 Overview of System 80+ Design

The following sections describe key System 80+ evolutionary design improvements, including design features that lead to higher plant availability and reduced costs, as well as improved severe accident prevention and mitigation.

2 KEY EVOLUTIONARY DESIGN CHANGES

The System 80+ Standard Plant Design meets the stringent design goals in the EPRI ALWR URD Volumes I and II. Specifically, the design complies with the EPRI goals of simplicity, improved reliability, improved accident prevention and mitigation, improved economics, and better man-machine interfaces Since the design represents an evolutionary advancement over current light water reactor designs, unproven features that could slow the licensing process and/or construction or could result in unreliable operation have been avoided. Accordingly, the design does not require any prototype testing

The System 80+ design also complies with the procedural requirements and criteria of NRC regulations including the Three Mile Island requirements codified in 10 CFR 50. In addition, the design addresses all applicable Unresolved Safety Issues and the medium- and high-priority Generic Safety Issues. Finally, a full Level III Probabilistic Safety Assessment (PSA) has been carried out for the design. The PSA was used as a guiding tool during the design process to produce a more robust design that minimizes the potential for core damage and moderates the severity of a severe accident should one occur. This PSA analysis has demonstrated that the System 80+ design meets the ALWR safety goals

Evolutionary features contributing to the significant safety improvements of the System 80+ design include.

- Increased reactor core thermal margin achieved by reducing the normal operating hot leg temperature, using ABB's advanced fuel design (TURBO™) and revising core monitoring methods.
- Use of a ring-forged reactor pressure vessel with improved material specification affording a low 60 year end-of life RT_{NDT}, virtually eliminating pressurized thermal shock concerns This feature also results in a significantly reduced number of welds (with resulting reduction in in-service inspection).

- Pressurizer volume is increased by 33% (relative to current generation operating reactors), providing more operating margin during plant transients.
- Secondary inventory in the steam generators is increased by 25%, increasing the time period until actuation of reactor protection/safeguards systems and until the steam generator would boil dry without provision of feedwater.
- Thermally-treated Inconel 690 tubing and enhanced maintenance features are used to extend the life of the steam generators, improve their reliability and decrease the potential need for plugging tubes over the life of the plant.
- N-16 gamma ray monitors, one per steam generator, have been incorporated in the steam lines to provide a sensitive and specific indication for primary coolant leakage through steam generators.
- A combustion turbine generator provides an alternate source of AC electrical power during loss-of-off site power and station blackout events.
- A dedicated Reactor Coolant Pump (RCP) Seal Injection System has been incorporated. With this system, three levels of protection (three sources of cooling water) exist, thereby essentially eliminating concerns for RCP seal failure and subsequent leakage during periods of prolonged electrical power loss.
- A Safety Injection System featuring four separate trains of equipment, direct vessel injection, elimination of the previous low-pressure safety injection function, and on-line test capability.
- A dedicated Emergency Feedwater System featuring two independent divisions (one for each steam generator), two full-capacity, seismically qualified storage tanks, diverse pumps (motor driven and steam turbine driven) in each division, and cavitating venturis to prevent pump runout and excessive flow to the ruptured steam generator.
- A Safety Depressurization System (SDS) has been added to provide rapid depressurization for severe accident mitigation and for back-up decay heat removal.
- An in-containment refueling water storage tank (IRWST) which acts as a quench tank for the SDS, avoids the need for safety injection recirculation switch over to the containment sump after a loss-of-coolant accident, and provides a guaranteed source of water for cavity flooding.
- The Cavity Flooding System allows for safety-grade flooding of the reactor cavity, manually initiated from the control room during a severe accident.
- A state-of-the-art main control room (Nuplex 80+) using modern human factors engineering techniques and off-the-shelf digital technology has been designed to facilitate the operators' duties during both normal and potential accident situations.
- Additional mechanical redundancy has been provided for the shutdown cooling and containment spray systems.
- Enhanced support systems (e.g., component cooling water) with increased redundancy and physical separation to match the reliability of the safety systems.
- A large free-volume containment provides additional margin against over-pressurization and ensures that global hydrogen concentration cannot reach detonable levels during an accident.
- A hydrogen igniter system, in conjunction with hydrogen re-combiners, ensures that hydrogen is controlled without global deflagrations.

3. MEASURES FOR COPING WITH SEVERE ACCIDENTS

The System 80+ Standard Plant Design is a more robust plant design that reduces the potential for core damage and moderates the severity of such an accident should one occur. Some of these features were summarized above, but they are discussed below specifically with respect to severe accident prevention and mitigation.

3.1. *Safety Injection System (SIS)*

The SIS (Figure 2) is designed to satisfy NRC regulatory requirements and, in the highly unlikely event of a loss-of-coolant-accident, inject borated water into the Reactor Coolant System. The System 80+ SIS incorporates a four-train safety injection configuration and an In-Containment Refueling Water Storage Tank (IRWST).

The System 80+ SIS utilizes four high pressure safety injection pumps to inject borated water directly into the Reactor Vessel. In addition, four safety injection tanks are provided. The SI pumps are normally aligned to the IRWST and a realignment for recirculation following a LOCA is not required. This system provides cooling to limit core damage and fission product release and ensures adequate shutdown margin.

The SIS also provides continuous long-term, post-accident cooling of the core by recirculation of borated water from the IRWST. Water, drawn from the IRWST by the SI pumps and the containment spray (CS) pumps, is injected into the reactor vessel and the containment. The SI water then enters the containment through the primary pipe break. This water and the CS water return through floor drains and the holdup volume tank to the IRWST. During this process, heat is removed from the IRWST water by the containment spray heat exchanger.

The SIS is capable of providing an alternate means of decay heat removal for those events beyond the licensing design basis in which the steam generators are not available. Decay heat removal, via feeding and bleeding of the RCS, would be accomplished using the SIS to feed, the Safety Depressurization System (SDS) to bleed, and the Shutdown Cooling System (SCS) for cooling of the IRWST water.

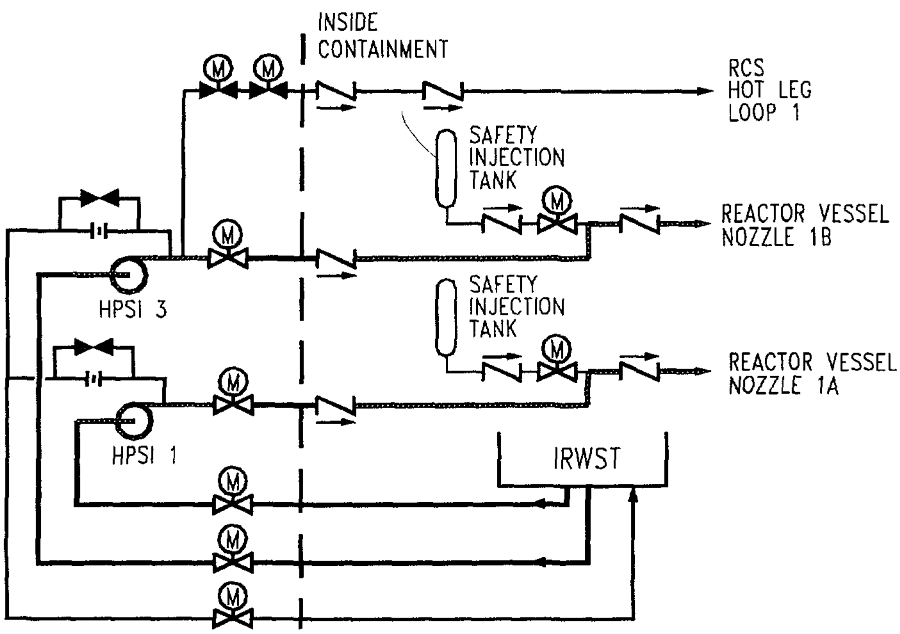

Figure 2. Safety Injection System

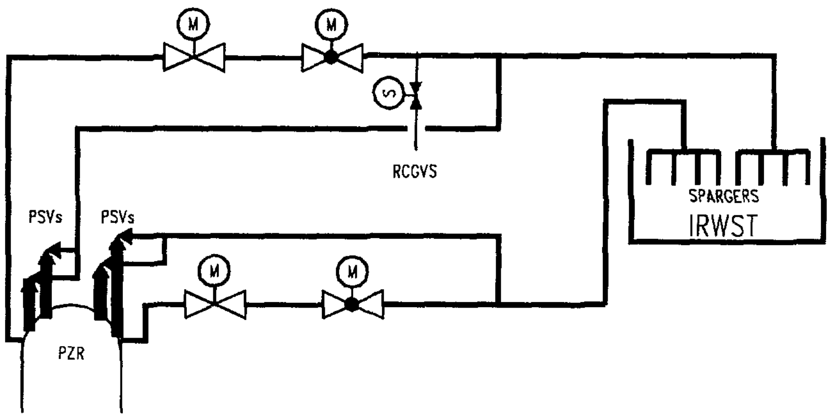

Figure 3. Safety Depressurization System

Pressure retaining components outside containment have a design pressure of at least 6.3 MPa (900 psig). This increased design pressure significantly reduces the chance of a large interfacing LOCA, even at full RCS pressure.

3.2. Safety Depressurization System

As a backup to the normal pressure control system and the Reactor Coolant Gas Vent System, the SDS (Figure 3) provides a safety-grade means of depressurizing the RCS. Together with the SIS and Shutdown Cooling System (SCS), the SDS is capable of providing an alternate means of decay heat removal for those events beyond the plant design basis in which the steam generators are not available.

In the context of severe accident prevention, the SDS performs the following functions:

- The Reactor Coolant Gas Vent function of the SDS provides a safety-grade means of venting non-condensable gases from the pressurizer and the reactor vessel upper head to the Reactor Drain Tank during post-accident conditions. In addition, the RCGV provides a safety-grade means to depressurize the RCS in the event that pressurizer Main Spray and Auxiliary Spray systems are unavailable.
- The Rapid Depressurization function, or bleed function, provides a manual means of quickly depressurizing the RCS when normal and emergency feedwater are unavailable to remove core decay heat through the steam generators.

In the event a high pressure meltdown scenario develops and the feed portion of feed and bleed cannot be established, the SDS can be used to depressurize the RCS to ensure that a High Pressure Melt Ejection event does not occur, thereby minimizing the potential for direct containment heating.

3.3. Containment Spray System (CSS)

The CSS for System 80+ is designed to maintain containment pres-sure and temperature with-in design limits in the unlikely event of design basis mass-energy releases to the containment atmos-phere.

The CSS (Figure 4) is a fully redundant two-train system. Two contain-ment spray pumps supply water through two heat exchangers to the upper region of the containment. Spray headers are used to provide a relatively uniform distribution of spray over the cross sectional area of the containment. The IRWST is used as the water source for the system. The Containment Spray Pumps can be manually aligned and used as residual heat removal pumps during SCS operation. Likewise, the SCS pumps can be manually aligned to perform the containment spray function.

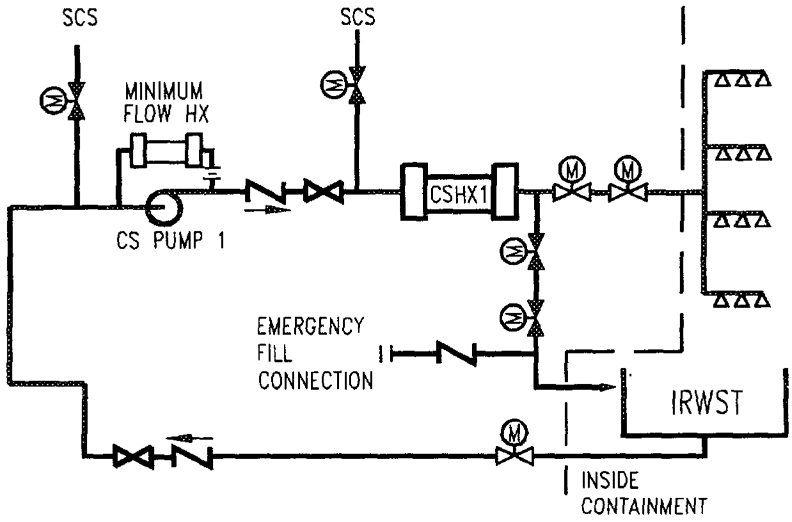

Figure 4. Containment Spray System

The CSS also provides a containment air cleanup function to reduce the concentration of fission products in the containment atmosphere after an accident. No spray additives are required. The CSS pumps and CSS heat exchangers can also be used as a backup to the SCS pumps and heat exchangers to provide cooling of the IRWST water during post-accident feed and bleed operations when the steam generators are not available to cool the RCS. The emergency fill connection enables water from an external source to be provided to the containment spray headers if the normal spray function is unavailable.

Pressure retaining components outside containment have a design pressure of at least 6.3 MPa (900 psig). This increased design pressure significantly reduces the chance of a large interfacing LOCA, even at full RCS pressure.

3.4. Emergency Feedwater System (EFWS)

The EFWS (Figure 5) is a dedicated four-train safety system (two trains per Division) that supplies feedwater to the steam generators for the removal of heat from the RCS in the event the main

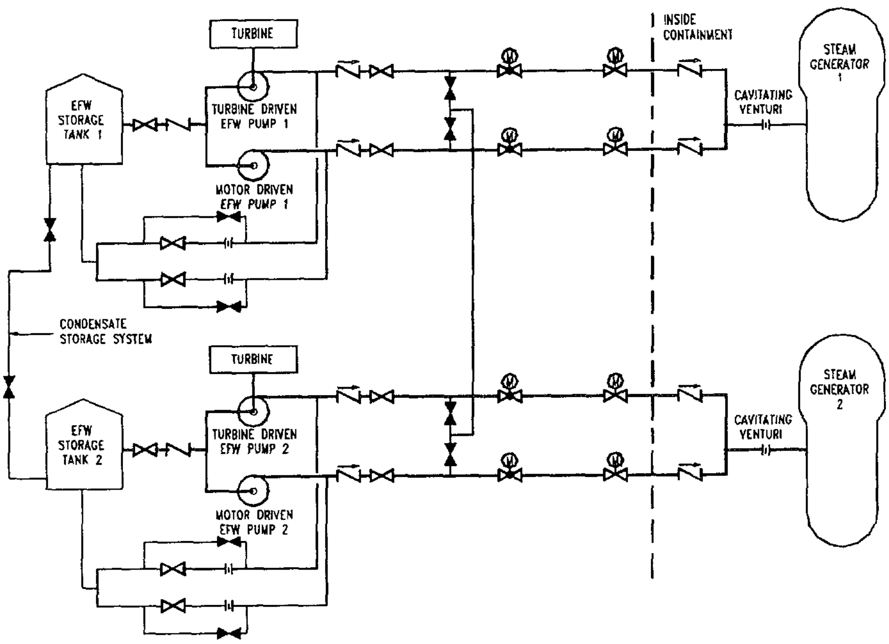

Figure 5. Emergency Feedwater System

feedwater system is unavailable following a transient or accident. The EFWS has no operating functions for normal operation.

The EFWS is configured into two separate mechanical divisions. Each division is aligned to feed its respective steam generator. Each division consists of one storage tank, one 100% capacity motor-driven pump subdivision, one 100% capacity steam-driven pump subdivision, valves, one cavitating venturi, and specified instrumentation. Each pump sub-division takes suction from its respective storage tank. The motor-driven subdivision and steam-driven subdivision are joined together inside containment to feed their respective steam generator through a common EFWS header. Each common EFWS header contains a cavitating venturi to restrict the maximum EFWS flow rate to each steam generator. The cavitating venturi restricts the magnitude of the two pump flow as well as the magnitude of individual pump runout flow to the steam generator, and eliminates the need for complicated isolation systems.

A cross-connection is provided between each EFWS storage tank so that either tank can supply either division of the EFWS. A line connected to a non-safety source of condensate is also provided that can be manually aligned for gravity feed to either of the storage tanks, should those tanks reach low level before Shutdown Cooling System entry conditions are reached. Pump discharge crossover piping is provided to enhance system versatility during long-term emergency modes, such that a single pump can feed both steam generators.

3.5. Shutdown Cooling System (SCS)

The SCS (Figure 6) is used to reduce the temperature of the reactor coolant, at a controlled rate, from 176.7° C (350° F) to a refueling temperature of 48.9° C (120° F) and to maintain the proper reactor coolant temperature during refueling. This system utilizes the shutdown cooling pumps to circulate the reactor coolant through two shutdown heat exchangers, returning it to the Reactor Coolant System. The Component Cooling Water System supplies cooling water for the shutdown cooling heat exchangers.

Additionally, the SCS is used in conjunction with the atmospheric dump valves (ADVs) and the Emergency Feedwater System to cool down the RCS following a small break LOCA. The SCS is also used subsequent to steam and feedwater line breaks, steam generator tube ruptures, and is used during plant startup prior to RCP restart to maintain flow through the core. After an accident, the SCS can be put into operation when the RCS pressure and temperature are below approximately 2.8 MPa (400 psia) and 193.3°C (380° F).

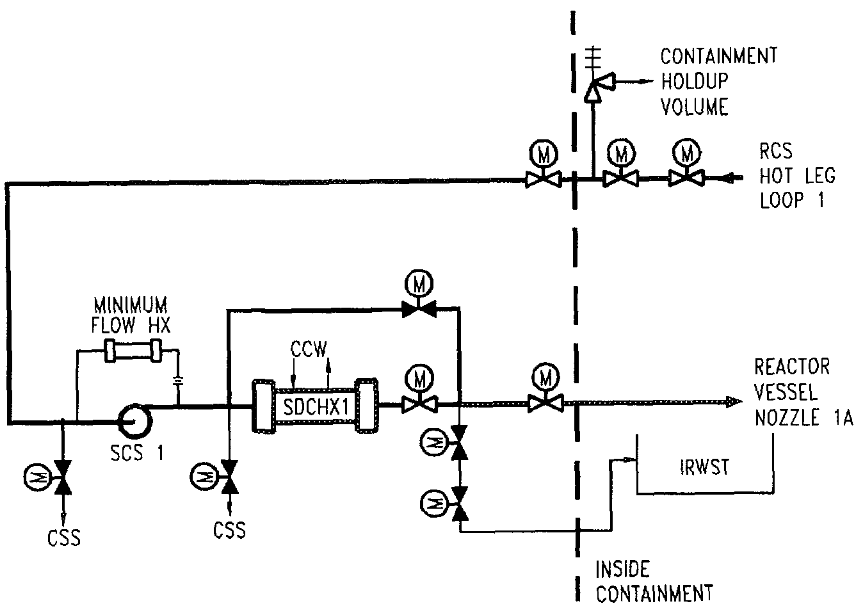

Figure 6. Shutdown Cooling System

The system has a design pressure of 63 MPa (900 psig). This higher system pressure provides for greater operational flexibility and significantly reduces the chance of a large interfacing system LOCA, even at full RCS pressure. The SCS pumps do not share functions with the SIS.

In addition to their normal, long-term decay heat removal function, the SCS pumps are designed to perform a backup residual heat removal function and cooling of the IRWST. In the backup residual heat removal mode of operation, the SCS is used (in conjunction with the Rapid Depressurization System) as a backup to the SIS to inject borated water into the reactor core. The CSS pumps can be used to backup the SCS pumps for improved decay heat removal capabilities. The SCS pumps can also be used as backups to the CSS pumps to perform IRWST cooling during "feed and bleed" operations (beyond design basis events). To further increase the reliability of the containment spray function, the containment spray headers are designed to accept spray flow from an external source of water via a "tee" connection to the spray line.

3.6. Safety Systems Configuration

The System 80+ plant is designed on the basis of two divisions of safety-related equipment which are each fully capable of achieving and maintaining a safe shutdown despite the complete loss of the redundant division's functionality. Outside of containment, the two redundant divisions of safety-related equipment are separated by an interdivisional barrier except for the control room and the remote shutdown room which are physically separated and electrically isolated from each other. The interdivisional barrier is a three hour fire rated barrier and a flood barrier. Heating, ventilation, and air conditioning and cooling water systems are designed to maintain divisional separation.

In areas inside the containment where equipment and cabling associated with safe shutdown equipment for the two divisions converge, an alternate capability is provided to achieve and maintain safe shutdown using systems and equipment spatially and electrically isolated from the convergent systems.

3.7. In-Containment Refueling Water Storage Tank (IRWST)

Sufficient borated water is stored in the IRWST to meet all post-accident safety injection pump and containment spray pump operation requirements. The IRWST eliminates the need for switching over from injection mode to recirculation mode during emergency core cooling operations and, therefore, eliminates failures associated with the switch-over in existing commercial nuclear power plants. In addition, the IRWST is the source of borated water for cavity flooding at the onset of a severe accident.

3.8. Two Emergency Diesels and Standby Combustion Turbine

Each of the two divisions of class 1E AC power is supplied with emergency standby power from an emergency diesel generator (DG). Each DG is provided with a dedicated 125 VDC battery. In addition to the two emergency DGs, the System 80+ design has an alternate standby onsite AC power source. This is a non-safety combustion turbine power source provided to cope with station blackout scenarios. The alternate power source is independent and diverse from the DGs.

3.9. Larger Pressurizer

The larger pressurizer volume in System 80+, as compared to the existing generation of commercial nuclear power plants, makes the plant response to transients slower with more limited pressure excursions. The larger volume also helps prevent emptying the pressurizer, uncovering the pressurizer heaters, opening the safety valves, and challenging the safety valves with two phase flow.

3.10. Larger Secondary Inventory in the Steam Generators

The larger secondary inventory in the System 80+ steam generators also makes the plant response to transients slower The increased downcomer volume and the 25% increase in steam generator inventory help reduce pressure and temperature fluctuations during transients and increase the time to dry out the steam generators The time required to deplete the secondary inventory of the steam generators is approximately 50% longer for System 80+ than for System 80

3.11. Containment Vessel

The design basis pressure for the containment is 365 kPa (53 psig) The analyses documented in CESSAR-DC demonstrate that pressures resulting from large break LOCAs or main steam line breaks within the containment will not exceed this design pressure Calculations also indicate that pressure limits determined in accordance with ASME Service Level C criteria range from 895 kPa (130 psig) at an average steel shell temperature of 143°C (290°F) to 826 kPa (120 psig) at a temperature of 232°C (450°)F The median ultimate failure pressure ranges from 1297 kPa (188 psia) at 66°C (150°F) to 1103 kPa (160 psia) at 232°C (450°F)

3.12. Secondary Containment

The secondary containment consists of the containment shield building and the annulus between the steel containment vessel and the shield building The containment shield building, which houses the containment vessel and safety-related equipment, is designed to provide biological shielding and external missile protection for the containment vessel and safety-related equipment In addition, the Annulus Ventilation System (AVS) provides a mechanism for substantially reducing or eliminating unfiltered fission product releases following design basis and severe accidents

3.13. Hydrogen Mitigation System (HMS)

The large System 80+ containment is designed to prevent hydrogen buildup by natural circulation and can passively accommodate a metal-water reaction of up to 75% of the core metal without exceeding a hydrogen concentration of 13% by volume While it is unlikely that explosion of hydrogen generated during a severe accident would fail the containment, an igniter-based HMS has been incorporated into the System 80+ design to provide added assurance that hydrogen concentrations will be maintained at non-detonable levels even during the most limiting severe accident The HMS is designed to accommodate the hydrogen production from 100% fuel clad metal-water reaction and maintain the average containment hydrogen concentration below the 10% limit for a degraded core accident The HMS also has hydrogen recombiners which ensure that the hydrogen concentration is maintained below 4% during a DBA LOCA

The HMS consists of two redundant groups of igniters, each group having independent and separate control, power, and igniter locations to ensure adequate coverage within the containment The igniters can be powered from any one of four sources normal offsite power, on-site emergency diesels, batteries, and the alternate AC power source (i e , the combustion turbine generator) The igniter enclosures provide protection from water jet impingement and minimize the temperature rise inside the igniter assembly The HMS is a manually actuated system designed to allow controlled burning of hydrogen at low concentrations to preclude hydrogen concentration build-up to detonable levels

3.14. Cavity Flooding System (CFS)

The CFS (Figure 7) provides a means of flooding the reactor cavity in the event of a severe accident for the purpose of cooling the core debris in the reactor cavity and scrubbing fission product releases The cavity flooding system is designed (in conjunction with the containment spray system) to provide an inexhaustible, continuous supply of water to quench the core debris The CFS is a manually

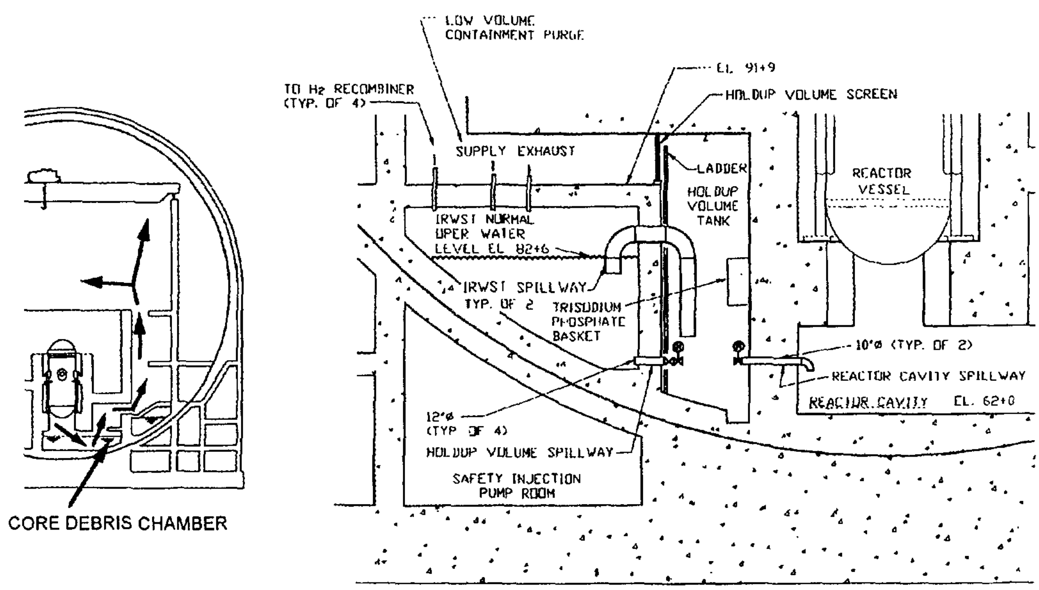

Figure 7. Reactor Cavity and Cavity Flooding System

actuated severe accident mitigation system. The components of the CFS include the In-Containment Refueling Water Storage Tank (IRWST), the Holdup Volume Tank (HVT), the reactor cavity, connecting piping, valves and associated power supplies. The CFS takes water from the IRWST and directs it to the reactor cavity. The water flows first into the HVT by way of four 30.5 cm (12 in) diameter HVT spillways and then into the reactor cavity by way of two 25.4 cm (10 in) diameter reactor cavity spillways.

3.15. Reactor Cavity Design

The System 80+ reactor cavity (Figure 7) is configured to promote retention of, and heat removal from, the core debris during a postulated severe accident, virtually eliminating the potential for significant DCH-induced containment loadings. The System 80+ cavity includes 906 m^3 (32,000 ft^3) of free volume. Large (and well vented) volumes such as those in System 80+ are not prone to significant pressurization resulting from vessel breach or the corium quench processes.

The vertical instrument shaft permits only limited gas venting, minimizing the potential for corium entrainment and discharge via this path. Corium not entering the shaft will be captured in the large debris retention chamber.

The design of the System 80+ reactor cavity ensures that actual venting to the upper containment either by the vertical instrument shaft or around the reactor vessel flange is small. Thus, steam exits the reactor cavity via a convoluted pathway above the top of the core debris chamber and through louvered vents under the refueling pool. In addition, System 80+ has an offset core debris chamber designed to de-entrain and trap debris ejected during a reactor vessel breach. The core debris chamber and the instrument shaft have been designed such that following a failure of the reactor vessel, high inertia core debris would de-entrain and collect in the debris chamber while the lower inertia steam/hydrogen/air mixture would negotiate a right angle turn and exit the reactor cavity via a convoluted vent path.

The reactor cavity is sized and configured to spread out the ejected core debris over the floor surface area during a postulated severe accident so as to meet the criterion of 0.02 m^2 per MWt of surface area below the vessel. The cavity floor and sump are constructed with a limestone aggregate concrete with a minimum $CaCO_3$ content of 17 percent. The flat floor area is free from obstructions to corium debris spreading. The minimum flat floor area for the reactor cavity is 64.4 m^2 (693 ft^2). Within the reactor cavity, the containment shell is protected from corium debris by a concrete basemat layer varying from 0.9 m (3 ft) to about 1.5 m (5 ft) thick. Assuming flooded cavity conditions, it is conservatively estimated that this thickness of concrete flooring will protect the containment shell from corium debris contact for more than 30 hrs at its thinnest point. In addition, the structural concrete below the containment pressure boundary provides at least 5 m (16 ft) of concrete which acts to prevent basemat melt-through for at least eight days.

Calculations show that the reactor vessel and the upper cavity could continue to be supported even if the entire lower cavity walls below the corbels were either eroded by corium attack or destroyed by a steam explosion. Reinforcing steel between the interface of adjacent walls with the upper cavity wall provide enough resistance through shear friction to provide this support without relying on support from the lower cavity wall. An Ex-Vessel Steam Explosion (EVSE) may occur when corium debris contacts a water pool. EVSE consequences with respect to containment integrity, however, are expected to be insignificant. EVSEs are not expected to be capable of damaging the reactor cavity structures required for support of the reactor vessel or RCS. All "in-cavity" structures that may be damaged by such explosions will be confined to non-load bearing structures and thus will not compromise containment integrity.

4. HIGH AVAILABILITY

One of the key objectives of the System 80+ design is to maximize the operational availability of the plant. An important component of this objective is incorporating the necessary features in the plant design to minimize the time required to conduct the Refueling and Maintenance Outages. The following two outage scenarios have been analyzed:

- Refueling-Only Outage
- Normal Refueling-Maintenance Outage

4.1. Refueling-Only Outage

The basis for the Refueling-Only outage are the requirements provided in the ALWR URD. This document states "*The ALWR fueling and refueling systems shall be designed such that the reactor can be refueled in a maximum of time of 17 days between the opening and closing of the plant output breaker. The Plant Designer shall provide a time-sequence analysis of all refueling tasks to verify that the 17 day requirement can be met. The tasks included in the refueling outage are only those tasks which must be performed each time the reactor is refueled.*"

The tasks that must be performed at each refueling outage include the following:

- Shutdown and cooling the Reactor Coolant System
- Disassembly of the Reactor Head Area
- Replacement of one-third of the fuel assemblies and in-core shuffle of the remaining two-thirds of the fuel assemblies
- In-core instrumentation assembly handling including replacement of approximately one-third of the assemblies
- Re-assembly of the Reactor Head Area
- Testing and heatup of the Reactor Coolant System

For the System 80+ design, the time required for the refueling-only outage is sixteen (16) days – meeting the 17-day requirement of the ALWR URD.

4.2. Normal Refueling and Maintenance Outage

The Normal Refueling-Maintenance Outage for System 80+ has a duration of 21 days from the time the generator is disconnected until the generator is reconnected to the grid. During this outage, maintenance and testing is performed on plant systems during the refueling. In addition to the tasks performed during the Refueling-Only Outage, the following tasks are included in the outage schedule:

- Main Turbine/Generator maintenance and inspection
- Steam Generator tube inspections
- Reactor Coolant Pump seal replacement (two pumps per outage)
- System inspection and maintenance
- Instrumentation calibrations
- In-service Inspections of systems

The time required to handle the fuel assemblies and verify the fuel assembly positions in the core is less than 80 hours and the total time for the complete refueling-maintenance outage is only twenty-one (21) days.

4.3. Operational Availability Summary

The System 80+ design includes the capability of either an 18-month or a 24-month refueling cycle. For an 18-month refueling cycle with only a refueling outage, the availability would be greater than 97%. With a refueling-maintenance outage, the availability would be greater than 96%. However, most refueling cycles also include unscheduled shutdowns and recent experience has shown that, for System 80+, 14 days of maintenance-only outage work could be expected during an 18-month fuel cycle. With a maintenance-only plus a refueling-only outage, the System 80+ availability would be greater than 94%.

For the past three years, the System 80 designs in operation have operated with an average availability of greater than 87% (based on World Nuclear Performance data) and some units have operated with availabilities well over 90%. Therefore, the availabilities stated above for the System 80+ design appear to be reasonable in view of the advanced design features discussed in this paper.

5. REDUCED COST

The savings resulting from the increased availability due to shorter refueling outages are $1,975,000 per year, based on a replacement power cost of $25,000 per hour. Other significant areas of savings are decreased staffing requirements due to reduced maintenance and inspection requirements, which are the bulk of the total savings. The decreased volume of potentially radioactive liquid and solid waste is another major area of savings. As most of the savings are associated with refueling outages and work performed concurrent with the refueling outages, the yearly savings are based on an eighteen-month fuel cycle.

The System 80+ design is based on the ALWR URD, which includes features for more economical maintenance operations. A key objective of the design has been to provide the necessary features to minimize the time and manpower required to perform operation and maintenance activities.

ABB, along with a US utility, has quantified the operation and maintenance cost savings associated with the System 80+ design. This study was performed by identifying the features included in the System 80+ design that result in operation and maintenance savings. The features were reviewed to determine the affects on different activities in the plant. A comparison was made between the System

Table 1 System 80+ Operations and Maintenance Cost Reductions

PLANT FEATURE	SAVINGS ($) PER CYCLE
Building Layout Improvements	$4,574,000
System Simplifications Due to Building Layout Improvements	$7,578,000
System Layout and Design for Maintenance Improvements	$6,016,000
Improved Radwaste Handling and Equipment Drains	$770,000
Improved Work Flow During Refueling	$1,947,000
TOTAL	$20,885,000 (18-month cycle)

80+ design and experience at currently operating nuclear plants to determine the reduction in activity times and materials. Experienced plant operators and maintenance staff participated in the study to ensure an accurate assessment of the System 80+ features. The specific features and corresponding savings are summarized in Table 1.

The savings in Table 1 correspond to an annual savings of $13,930,000 and savings of $835,800,000 over the 60 year life of a System 80+ plant. These savings are annual savings per unit and would double for a two-unit site.

6. CONCLUSION – IMPROVED SAFETY AND ECONOMICS

Design advancements incorporated into the System 80+ design have resulted in a lower total core damage frequencies (CDF) and lower offsite large release frequencies for hypothetical severe accidents. Using methods reviewed and approved by the NRC, the total CDF was decreased to <3.5E-6 events/year, easily meeting the ALWR goal of 1.0E-5 events/year. Similarly, the large release frequency was reduced to <5.3E-8 events/year, well below the ALWR goal of 1.0E-6 events/year. In addition, for an individual at the site boundary, the mean individual Total Effective Dose Equivalent weighted over all core damage events is only 0.0052 Sv.

In view of this improved safety and the operations and maintenance benefits summarized in the previous sections, the System 80+ design meets the requirement that ALWRs be more economical as well as safer.

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BWR 90 & BWR 90+ — TWO ADVANCED BWR DESIGN GENERATIONS FROM ABB

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Abstract

ABB has two evolutionary advanced light water reactors available today - the BWR 90 boiling water reactor and the System 80+ pressurised water reactor. The BWR 90 is based on the design, construction, commissioning and operation of the BWR 75 plants. The operation experience of the six plants of this advanced design has been very good. The average annual energy availability is above 90%, and total power generation costs have been low. When developing the BWR 90 specific changes were introduced to a reference design, to adapt to technological progress, new safety requirements and to achieve cost savings. The thermal power rating of BWR 90 is 3800 MWth (providing a nominal 1374 MWe net), slightly higher than that of the reference plant. ABB Atom has taken advantage of margins gained using a new generation of its SVEA fuel to attain this power rating without major design modifications. The BWR 90 design was completed and offered to the TVO utility in Finland in 1991, as one of the contenders for the fifth Finnish nuclear power plant project. Hence, the design is available today for deployment in new plant projects. Utility views were incorporated through co-operation with the Finnish utility TVO, owner and operator of the two Olkiluoto plants of BWR 75 design. A review against the European Utility Requirement (EUR) set of requirements has been performed, since the design, in 1997, was selected by the EUR Steering Committee to be the first BWR to be evaluated against the EUR documents. The review work was completed in 1998. It will be the subject of an *"EUR Volume 3 Subset for BWR 90"* document. ABB is continuing its BWR development work with an "evolutionary" design called BWR 90+, which aims at developing the BWR as a competitive option for the anticipated revival of the market for new nuclear plants beyond the turn of the century, as well as feeding ideas and inputs to the continuous modernisation efforts at operating plants. The development is performed by ABB Atom together with TVO. Swedish BWR operators have also joined the project.

1. INTRODUCTION

Today, ABB has a modern BWR design, the BWR 90, that has already been offered commercially. This design was selected by the European Utility Requirements (EUR) group to be reviewed for compliance with its set of requirements. ABB is continuing its BWR development work, however, with focus on the 21st century, on a new design called BWR 90+ that offers reduced costs and significant safety improvements. The work aims at providing an economical alternative based on evolutionary development of the earlier advanced BWR design. The design goal is a 1500 MWe plant that can be built in less than 1500 days.

A second purpose of the development activities is to provide input to improvements and modernisation of earlier generations of nuclear power plants.

2. DEVELOPMENT BASED ON SUCCESSFUL OPERATION EXPERIENCE

ABB Atom has a long tradition of plant and system development activities related to nuclear power plants. Its first BWR unit, Oskarshamn 1 nuclear power plant in Sweden was taken into operation in 1972, - developed and built without reliance on licenses. This design incorporated a number of advanced features such as a pre-stressed concrete containment, fine-motion control rods and a passive isolation condenser. Subsequent plants were designed much along the same lines as Oskarshamn 1, with step-wise improvements.

A major step forward came with the advanced BWR 75 design, which is characterised by use of internal recirculation pumps, fine-motion control rod drives, four independent and physically separated trains of engineered safety systems, and a pre-stressed slip-formed containment. Six nuclear power plants of this design are in operation in Sweden and Finland. The accumulated successful operation experience of these plants amounts to almost 100 reactor-years, and demonstrates the capability of being operated at high energy

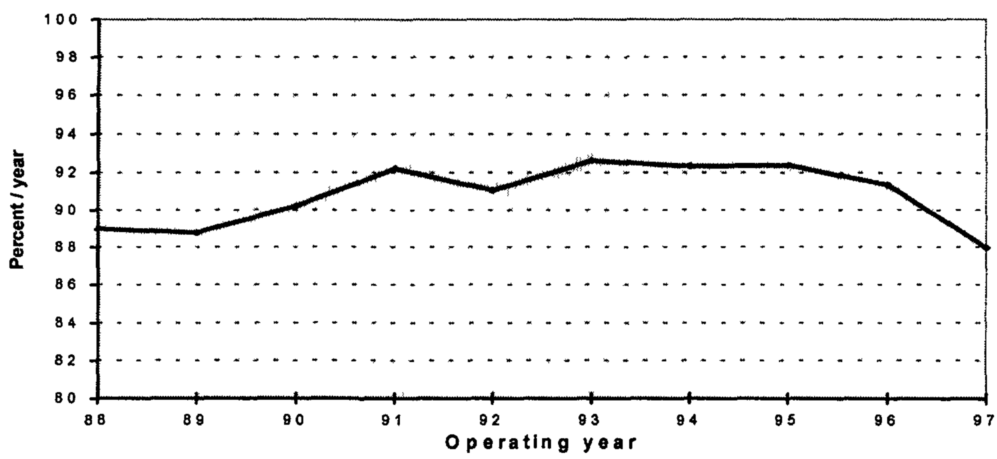

FIG. 1. Annual energy availability factors for the BWR 75 plants.

availability factors (figure 1). The total electricity generation costs have been low, as demonstrated by the published production costs for the Forsmark 1, 2, and 3 plants during the last decade (figure 2).

3. THE BWR 90 DESIGN FOR THE 1990s

The most recent BWR design of ABB, the BWR 90, was offered commercially to Finland in 1991, as one of the contenders for the fifth nuclear power plant project in Finland.

The BWR 90 design is based on the experience from design, construction, commissioning and operation of the BWR 75 plants in Finland and Sweden. Specific changes were introduced to an established reference design, that of the Forsmark 3 and Oskarshamn 3 units. Modifications were made to adapt to technological progress, new safety requirements and to achieve cost savings. An efficient feedback of operation experience was provided by a co-operation with the Finnish utility Teollisuuden Voima Oy (TVO) that operates the Olkiluoto 1 and 2 plants of BWR 75 design in Finland. The operating experience of these units has been very satisfactory; e.g., the average capacity factor over the last ten years is 93,2 %.

The design is characterised by the use of internal recirculation pumps, fine-motion control rod drives, and comprehensive physical separation of the four-train safety systems, basically in the same way as in its predecessor, the BWR 75. The thermal power rating of the base version is 3,800 MWth, somewhat higher than that of the reference plant, supplemented by a smaller unit of 3,300 MWth. The BWR 90 takes advantage of

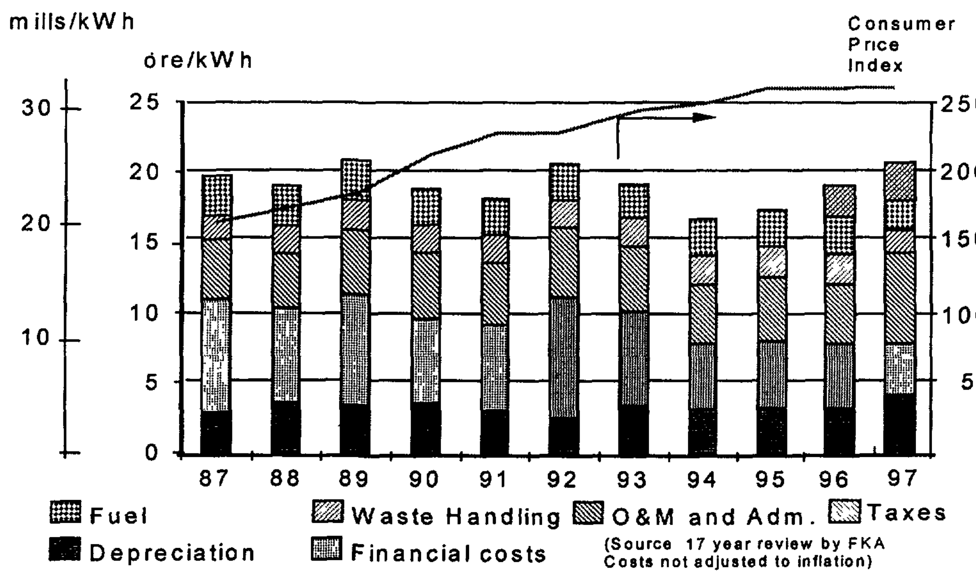

FIG. 2. Generation costs for the three BWR 75s at Forsmark.

margins gained by BWR fuel development results, e.g., the introduction of a fuel with four sub-assemblies of 5x5 fuel rod array, offering increased heat transfer capability and thereby permitting power increase.

In 1997, the Steering Committee of the European Utility Requirements (EUR) group, selected the BWR 90 design to be the first BWR design to be evaluated and reviewed for compliance with the EUR document. This work that is scheduled for completion by the end of 1998, has comprised a detailed assessment against the overall requirements - in Volumes 1 and 2 of the EUR document. The results will be incorporated into a "Volume 3 Subset for the BWR 90 design" document. As a matter of fact, the effort also serves to demonstrate the general applicability of the EUR document to BWR designs. The review has shown that BWR 90 meets most of the requirements; deviations mainly refer to technical details.

4. SOME BWR 90 HIGHLIGHTS

A main emphasis in the development work has been to maintain "proven design" features, unless changes would yield improvements and simplifications. In line with this philosophy, the reactor design has changed very little.

The core design is closely the same as in previous ABB BWRs. Advanced utilisation of burnable absorber material (Gd_2O_3) in the fuel made it possible to achieve good axial and radial power distribution with low peaking factors. The SVEA-100 fuel with its thinner fuel rods yielded an increased heat transfer surface and improved operating margins. For the BWR 90, the improved power peaking factors and the increased margins were taken into account to raise the power level of the reactor.

In other respects, the core design and arrangement are not changed. The control rod and control rod drives are of the well-proven ABB design with a grey-tipped solid steel blade cruciform rod; the neutron-absorbing material is filled into horizontal bores - boron carbide along the blade length and hafnium at the blade tip.

The steam separator units have been improved - and the steam dryers as well - in order to ensure low moisture content in the steam at the increased power output; the basic arrangement is just the same as in previous plants.

The reactor pressure vessel has been modified slightly. The cylindrical portion is made up of cylindrical forgings in the same way as in the Forsmark 3 and Oskarshamn 3 plants; this eliminates the longitudinal welds. The bottom portion is redesigned in such a way that large sections can be made by forging; the number of welds is reduced significantly. This reduction in number of welds is important for the plant operation since it reduces the amount of in-service inspection to be carried out during the refuelling outage.

4.1 Reactor coolant system

The recirculation system is based on the use of internal glandless pumps driven by wet asynchronous motors, supplied with "variable frequency & variable voltage" power from individual frequency converters. This type of pump has been operating reliably in ABB BWR plants (for more than four million operating hours) since 1978. The internal pumps provide means for rapid and accurate power control and are advantageous for load following purposes; such internal pumps are now employed also by other BWR vendors, in the ABWR plants.

4.2 Safety systems

The engineered safety systems in BWR 90 are characterised by a consistent division into four redundant and physically separated subsystems. This concept was introduced in Olkiluoto 1 and 2 and further developed in Forsmark 3 and Oskarshamn 3, and has been reconfirmed as an optimal arrangement with respect to safety, layout and maintainability. For the emergency cooling systems, this implies that each of the four subsystems is located in its own bay, adjacent to the reactor containment and surrounded by thick concrete walls. The physical separation is maintained all the way to the ultimate heat sink. The individual compartments for safety-related subsystems and components constitute separate fire areas and fire cells.

The safety-related auxiliary electrical power supply equipment is in the same way divided into four independent and physically separated parts, and the reactor protection system operates on a 2-out-of-4 logic for signal transmission and actuation.

The four safety-related, standby power diesel generators with their auxiliary equipment are installed in two diesel buildings. They are located at opposite sides of the reactor building, which provides a high degree of physical protection. The diesel buildings house also pumps and heat exchangers for safety-related cooling systems, as well as safety-related auxiliary power supply and control equipment.

The capacities of the emergency core cooling systems suffice to provide water under all postulated pipe break conditions without any uncovery of the fuel. This statement is also valid assuming that only two of the four redundant subsystems are operable; the postulated loss-of-coolant conditions include a hypothetical 80 cm^2 leak at the bottom of the reactor vessel. In this context, it can be noted that the capacity of the low-pressure coolant injection pumps has been reduced for BWR 90, following comprehensive core cooling analyses. As a secondary effect, it has been possible to simplify the auxiliary power supply systems.

BWR 90 predominantly relies on active systems and components to realise safety functions; diversity is mainly found in the use of different process parameters and backup systems. Examples are the scram backup circuit of the reactor protection system, the use of diverse types of pilot valves for the pressure relief, and the filtered containment venting system for residual heat removal.

Another feature along this line is the traditional ABB BWR control rod drives system that incorporates diversified means of control rod actuation and insertion, by hydraulic pressure and by electrical motor. Together with a generous reactor pressure relief capacity, and combined with a capability of rapid recirculation flow rate reduction (by pump runback), it provides an efficient ATWS (Anticipated Transient Without Scram) countermeasure.

4.3 Reactor auxiliary systems

The main development objective related to the reactor auxiliary systems was to evaluate possible simplification of their design in order to achieve cost reductions and more straight-forward operation. An example on this approach is given by the reactor water cleanup system (RWCU). In previous plants, a portion of the full-power reactor coolant flow rate was continuously passed through the RWCU filters, and a forced flow mode (at a doubled flow rate) was initiated when needed. In the BWR 90, the RWCU operation is controlled by the reactor water chemistry conditions; during normal full power operation cleanup need is limited, and only a small flow of reactor water is passed through the RWCU. However, when measurements show a need, the RWCU is taken into operation at full capacity. This reduces heat losses, and therefore yields "cost reductions".

4.4 Containment and auxiliary buildings

The plant and buildings of the BWR 90 are laid out and designed to satisfy aspects of safety, maintenance and communication in a balanced way. The layout is strongly influenced by safety requirements, in particular the physical separation of safety-related equipment.

The general arrangement of the buildings (cf. Figure 3) is characterised by a division into a nuclear, safety-related part of the plant, containing the reactor building, the diesel buildings and the control building, and a more "conventional" part that is "separated" from the former by a wide communication area. The "conventional" part contains the turbo-generator and auxiliary systems of the plant. This arrangement is advantageous when building the plant as well as during plant operation, since the conventional part does not interfere with the nuclear part.

Compared with previous plants, building volumes have been substantially reduced, yielding a significant cost reduction. Nevertheless, BWR 90, like previous plants, is characterised by a fairly spacious layout. This facilitates access to components and is a key to low occupational exposure.

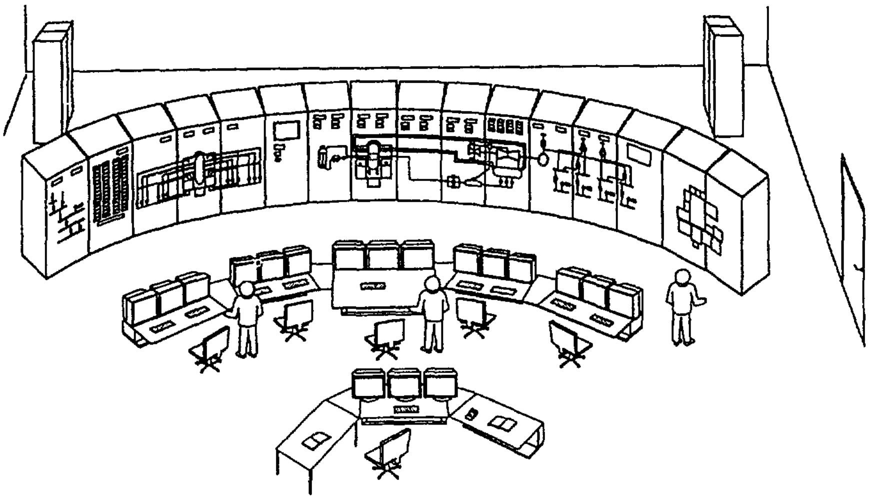

FIG. 4. BWR 90 Control Room Arrangement.

The BWR 90 pressure-suppression containment consists of a cylindrical pre-stressed concrete structure with an embedded steel liner - as in all previous ABB BWR plants. The containment vessel, including the pressure-suppression system and other internal structural parts as well as the pools above the containment, forms a monolithic unit and is statically free from the surrounding reactor building.

At the time of the design review, regulatory developments indicated a need to strengthen the capability of the reactor containment to withstand the effects of a core melt accident. Today, such requirements are codified in several countries, e.g., Finland and Sweden. The essential features of the BWR 90 containment to achieve enhanced environmental safety including protection during a degraded core accident, are:

- The blow-down of steam to the suppression pool passes through vertical concrete pathways to horizontal openings between drywell and wetwell.
- The relief pipes from the safety/relief valves are drawn into the suppression pool via the lower drywell rather than penetrating the drywell-wetwell intermediate floor.
- A pool is provided at the bottom section of the lower drywell for the purpose of collecting and confining fuel melt debris. The pool is permanently filled with water to enhance passive safety.

In addition, the containment vessel can be vented to the stack through a filter system, installed in the reactor building, similar to the filtered venting systems installed at all nuclear power plants in Sweden. These arrangements improve the reliability of the pressure-suppression system and reduce the probability of containment leakage during a severe accident.

4.5 Electrical distribution and support systems

The electrical power systems for safety-related objects are strictly divided into four separated subdivisions - a principle that is implemented in the operating BWR 75 plants and maintained in the BWR 90.

For the ordinary power distribution, some simplifications have been introduced. In previous plant designs, voltage stability considerations, limiting the ratio between direct-starting motor loads and available short circuit power on each busbar, represented a design constraint. In the BWR 90, the ratings of some of the major plant loads have been reduced by design changes in process systems. The condensate and main feedwater pumps in the turbine plant have been provided with static power supply converters, which result in reduced inrush currents during pump startups, i.e., the requirements on available short circuit power on the busbar system are significantly reduced. Switchgear components, with higher short circuit ratings, are also now available, and consequently a significant simplification of the structure of the auxiliary power supply systems has been made possible.

Another visible feature is that the number of distribution voltage levels have been reduced. As an example, it can be noted that DC distributions at several voltage levels for power supply to control equipment has been replaced by power supply from the battery-backed AC distribution, using distributed AC/DC converters for the supply to the various types of equipment, when needed. These simplifications of the electric power systems will of course also have a significant influence on the amount of maintenance work; a substantial reduction is anticipated.

4.6 Instrumentation and control systems.

A key to modern process control and communication applied to the BWR 90 is the use of control and instrumentation systems based on micro-computers. Process communication with the control room is realised by means of distributed functional processors. These in turn interact via serial communication links with a number of object-oriented process interface units. Thus, the protection and control system configuration is characterised by decentralisation and the use of object-oriented intelligence. The arrangement satisfies the requirements of redundancy and physical separation. It includes intelligent self-monitoring of protective circuits.

The use of serial communication links guarantees interference-free performance and reduces cabling. Standardisation of the object-oriented circuits minimises maintenance and the necessary stock of spare parts. The arrangement will also tend to improve availability, since components can be replaced quickly and simply. An important aspect is that the software is also standardised to simple program functions. This makes it easy even for non-computer specialists to handle the systems, and it facilitates implementation of new micro-computer generations.

Video display units (VDUs), keyboards, and display maps are used consistently to facilitate the man-machine communication in the control room. The main control room (cf. Figure 4) contains several work positions, each equipped with three VDUs. Typically, one VDU will display a total view of the process in interest, another will provide a list of alarms, and a third VDU will display a diagram with sufficient detail to facilitate operator action. This arrangement is supplemented with a special overview panel, visible to all operators in the control room. The overview shows the main process in the form of a flow diagram and indicates the status (normal, disturbed or failed) of various plant functions by conventional instruments and computer-based displays.

The main computer has the task of collecting information from the process control systems, and it communicates with the distributed micro-computers via serial links. The main computer compiles information and generates reports, such as daily/weekly operation reports, reports of periodic testing, actual status reports, and disturbance reports. During normal plant operation, the main computer will present occurrences on a special VDU display located between the "overview" panel for the plant and that for the safety systems and functions.

Further information on the BWR 90 design is found in the IAEA Status report on Advanced Light Water Reactor Designs, IAEA-TECDOC-968 [6].

5. THE BWR 90+ DESIGN

ABB is continuing its BWR development with work on a new advanced design, the BWR 90+. The aim of the continued programme is to maintain and develop the BWR as a competitive option for a reviving market, primarily in Europe, beyond the turn of the century, as well as feeding ideas and inputs to the continuous modernisation efforts at operating plants. In essence, the programme is firmly based on evolutionary development of the company's previous, advanced BWR designs.

The main objectives of the project focus on anticipated utility needs in the 21st century; the new design should offer reliable power generation at reduced construction and operation costs and incorporate significant safety improvements. The on-going review of the BWR 90 against the EUR is of importance for the BWR 90+ development work, and observed findings are specifically addressed in the design.

The development work is conducted in co-operation with TVO, which feeds adequate operation experiences into the project and performs analyses of different design alternatives regarding safety aspects, including mitigation of severe accidents. Swedish BWR operators have also recently joined the project with focus on getting ideas and inputs for modernisation and improvement works at their existing nuclear power plants.

TVO started a large modernisation programme for its Olkiluoto plants in 1994, including safety improvements and power upratings. The involvement in the present BWR 90+ development work is part of the utility's long-term strategy. They consider it necessary to keep informed about development trends, and to be prepared for construction of a new unit by the turn of the century, to be able to meet the increased demand for electricity generating capacity in Finland. TVO has also initiated an environmental study for a new nuclear power plant unit at the Olkiluoto site.

5.1 Design and performance goals

Economic competitivity is of paramount importance for a new nuclear power plant design. The BWR 90+ design work aims at developing a plant with: - reduced investment cost, - short construction time, - high energy availability, - short refuelling outages, - low operation and maintenance costs, - low fuel cycle cost, as well as - low waste management and decommissioning costs.

With respect to flexibility and reliability the governing design guideline is: "Proven system design and components are to be adopted to ensure reliable electricity production, and moderate development steps are introduced only when bringing improvements." As a result, most of the fundamental design features from the previous designs with respect to the energy production capability and reliability will be incorporated also in the BWR 90+ design.

Some specific design and performance goals of the BWR 90+ development project are:

(a) Plant nominal power output; 1500 MWe
(b) Construction time; less than 1500 days
(c) Energy availability; higher than 90 %
(d) Refuelling outage; 15-20 days/year.

An increased plant size will generally yield a reduction in the cost per unit of produced energy. Therefore, the plant size has been increased from 1350 MWe for the BWR 90 design to 1500 MWe for the BWR 90+ design. Results from the fuel development, e.g. the SVEA Optima, are fully incorporated in the design.

Interest during construction represents a large portion of the cost for a new nuclear power plant, and a shortened construction time will thus be cost effective. In earlier projects, ABB used a number of methods to reduce construction times. As an example, in the Oskarshamn 3 project the containment liner with associated reinforcement arrangement was built as a very large module in parallel with the erection of the basement structure. The module was then slid into the basement and placed in the correct position.

The BWR 90+ containment design and the reactor service room layout allow extensive use of slipforming and modular construction methods. The reactor building and the auxiliary buildings will be built with an extensive use of prefabrication. The use of new construction methods has been considered in the layout. As a result, the construction time, from pouring of first concrete to start demonstration run, has been estimated to be less than 1500 days.

A high energy availability will contribute significantly to a low energy production cost. The BWR 90+ design incorporates the important design features from the BWR 75 plants that have demonstrated an excellent operating record; over the last decade, corresponding to 60 reactor-years of operation, they have reached an average annual energy availability above 90 %. Therefore, the BWR 90+ plant is expected to attain similar results, and meet the EUR requirement of 87 % over 40 years of operation. Besides, an efficient feedback of operating experience from TVO, and the Swedish utilities, should bring improvements regarding operation and maintenance aspects.

TVO has demonstrated the feasibility of very short refuelling outages, down to ten days, in the 710 MWe Olkiluoto 1 and 2 nuclear power plants. For the larger 1500 MWe BWR 90+ design, refuelling outages in the order of 15-20 days are seen as a realistic target.

5.2 New evolution and safety requirements

The BWR 90+ design builds firmly on proven design, but considers and adopts new developments including new technology, digitised control equipment, and passive features and functions, as well as features that yield improved severe accident mitigation. The design principles are based on generally established international codes and standards. In addition, specific attention is paid to the requirements of STUK, the regulatory body of Finland. The EUR documents and the EPRI URD[1] are also considered.

The safety requirements applied in the design lead to use of redundancy and diversification in addition to physical separation to ensure independence. The diversification includes use of passive systems. The design follows the "defence-in-depth" principles, aiming at ensuring:

(1) low frequency of disturbances;
(2) that disturbances are controlled and do not develop into more severe events;
(3) that accidents are mitigated and do not develop into a core melt accident; and
(4) that the effects on the surroundings of a core melt accident remain acceptable.

The most recent edition of the STUK guides[2] addresses the need for a very high safety level and calls for improved severe accident mitigation, and limited accident consequences. Some examples are:

(a) The plant shall be designed so that no release of radioactivity will occur during the first period after a severe accident, even if all easily oxidising materials in the reactor core react with water.
(b) The design should include a containment venting system, but containment venting shall not be the primary way to reduce a containment overpressure in a severe accident situation.
(c) The relief valves shall be activated only temporarily (for brief discharges) during an anticipated transient to control the reactor pressure.
(d) The plant shall be designed to prevent release of radioactive matter in case of a possible accident, including a LOCA, during shutdown conditions, e.g., resulting from human errors during refuelling.
(e) Provisions shall be made to ensure decay heat removal in the event of loss of the ultimate heat sink normally used.

In order to meet these demands, a number of changes are introduced in the BWR 90+ design, compared with previous designs, including an improved containment design, introduction of passive systems and ECCS modifications. Design measures to cope with a "degraded core" accident have been incorporated in the containment design by provision of a core catcher arrangement and filtered venting for the containment.

The systematic division of the engineered safety systems into four independent and physically separated sub-systems (trains) in the predecessor yields benefits with respect to both safety and maintenance. Two-out-of-four trains provide the safety function that is required, and therefore maintenance may be performed during operation of the plant.

The control and instrumentation systems will be based on digital programmable equipment – with a computer-based control room, where the man-machine communication builds on video display units, keyboards and large display overviews.

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[1] EPRI URD – Utility Requirements Document of Electric Power Research Institute, USA.

[2] Guide YVL 1.0 – Safety Criteria for Design of Nuclear Power Plants.

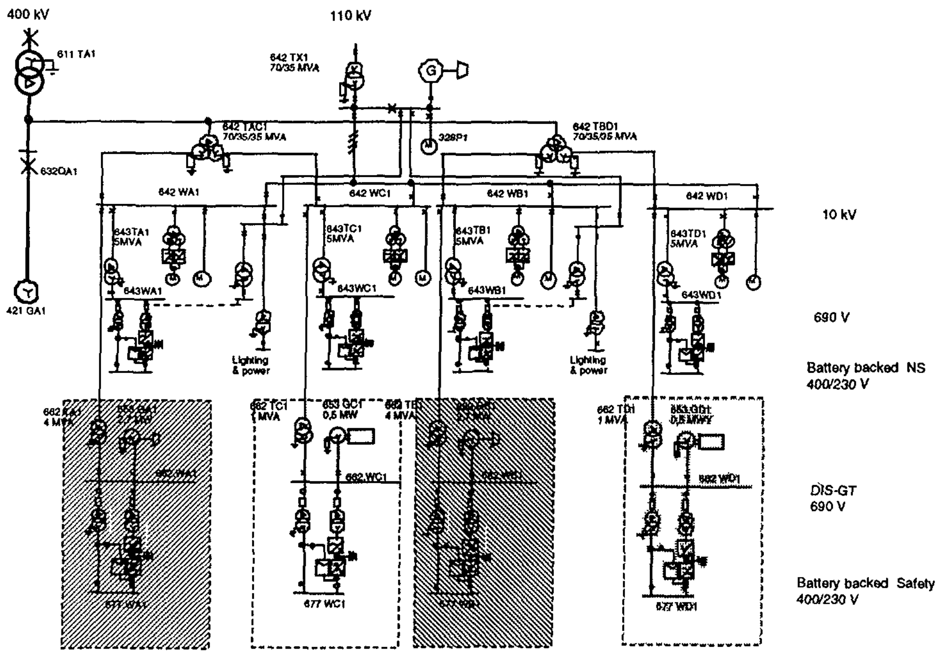

FIG. 5 BWR 90+ - Single line diagram

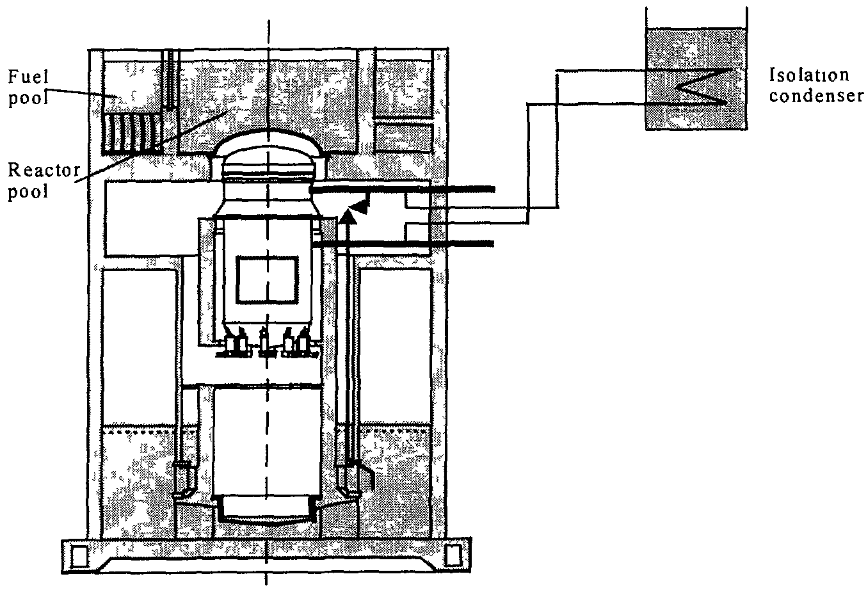

FIG. 6. Primary containment arrangement

The emergency power supply system (cf. Figure 5) is diversified by use of diesel generators and gas turbine driven generators. The number of distribution voltage levels has been reduced. All distributions are by AC, and local AC/DC converters are used where needed. The simplifications introduced will reduce maintenance work considerably and less space and cables are needed.

5.3 Improved containment design

The primary containment (cf. Figure 6) is characterised by robust design principles. During normal operation, the containment is inerted by nitrogen gas, thereby eliminating the risk of fires during operation and the risk for hydrogen explosions in case of postulated core melt accidents.

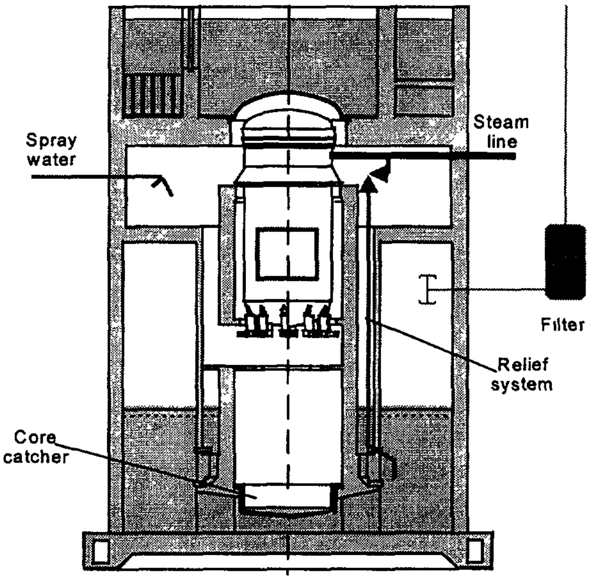

FIG. 7. Severe accident mitigation features.

TABLE 1. VOLUME COMPARISONS WITH PREVIOUS PLANT DESIGNS

	Power MW_{th}	Wetwell water volume m^3	Ratio DW/WW
BWR 90+	4250	6660	1,02
OL1/OL2	2500	2700	1,5
BWR 90	3800	3310	2,2
F3/O3	3300	3310	2,06

Except for vacuum breakers, all pipe connections between the drywell and wetwell have been eliminated. The number and size of the vacuum breakers have been reduced. The wetwell, including the partitioning floor, is provided with a leak-tight liner in stainless steel. This design minimises the potential for drywell - wetwell bypasses.

A dry core catcher (cf. Figure 7) is arranged beneath the reactor pressure vessel; its steel structure is submerged into the containment pool. In case of a severe accident, involving core melt and penetration of the reactor pressure vessel, the molten core will be collected in the core catcher, which will be cooled by the surrounding water. The containment structure is protected against the direct impacts of the molten material and does not serve as the primary barrier for a core melt. The containment proper will serve as an inherently passive system ensuring that no releases of radioactivity to the environment will occur during the first period after a severe accident with a molten core. The improved design implies a reduced risk for steam explosions, and a released molten core will be cooled in a passive way by the containment pool water. In addition, the core catcher arrangement eliminates the potential for major core-concrete interaction.

The wetwell gas compression chamber volume and the pool water volume have been increased compared with previous designs, as shown in Table 1. The improved design will accommodate the pressure build-up that may occur from hydrogen generation from all zirconium in the core for one day without activation of the cooling, overpressure protection, and support systems. Activation of active cooling systems, as well as spraying water into the drywell, will cool the containment structure, reduce the containment pressure and, in turn, prevent releases to the surrounding. The filtered venting system can be used to reduce the containment pressure in the long term without concerns for significant off-site consequences.

In the BWR 90+ design, there are no openings or pipe and cable penetrations from the lowest part of the drywell. The top of the core is located below the level of the upper drywell (or partitioning) floor. In the hypothetical case of a LOCA induced by human errors during plant shut-down and refuelling operations, the water volume in the pools above the reactor will suffice for filling the drywell volume to above the partitioning floor, and consequently, this design implies that the core will remain flooded without human action or safety system actuation.

Nuclear power plant containment structures can be characterised as pressure vessels. Logically, and in similarity with the code requirements for pressure relief equipment on other pressure vessels, the containments of all Nordic BWRs, as well as the BWR 90+ design, are equipped with pressure relief equipment. These consist of a safety rupture disk connected to the wetwell gas atmosphere and a parallel valve that can be opened manually. The drywell atmosphere will first be filtered through the containment pool water. In the longer term, the atmosphere from the wetwell can then be released via a filtered vent system. In addition, by sprinkling water to the drywell atmosphere the pressure can be reduced, the concrete structure cooled and activity can be transferred to the containment pool.

The improved containment design – with the pools on its top - is fully adapted to construction by means of slip-forming methods; the peripheral walls of the pools are made as integral parts of the containment wall structure. Combined with an extensive use of modular building technique, this reduces the construction time and costs.

The main features of the improved containment design are:

(a) Reduced construction time and costs.
(b) Minimised probability for drywell - wetwell bypass.
(c) Core remains covered by water if loss of coolant accident occurs during refuelling.
(d) Passive core melt retention and cooling inside containment; no releases within one day in the event of a core melt accident.
(e) Containment structure is protected against core melt impact by core catcher arrangement.
(f) A dry core catcher reduces the probability for steam explosions.
(g) Core concrete interaction is negligible.
(h) Increased volumes cope with pressure build-up from hydrogen generation at core melt accident.
(i) Nitrogen gas inertion allows cooling and pressure reduction by water spraying without risk for hydrogen explosions.
(j) Ultimate overpressure protection by filtered containment venting.

5.4 Decay heat removal and safety systems

The safety system configuration in the BWR 90+ design is characterised by a mixture of diversification, redundancy and separation and the use of passive systems. The four times 50 % approach introduced in the

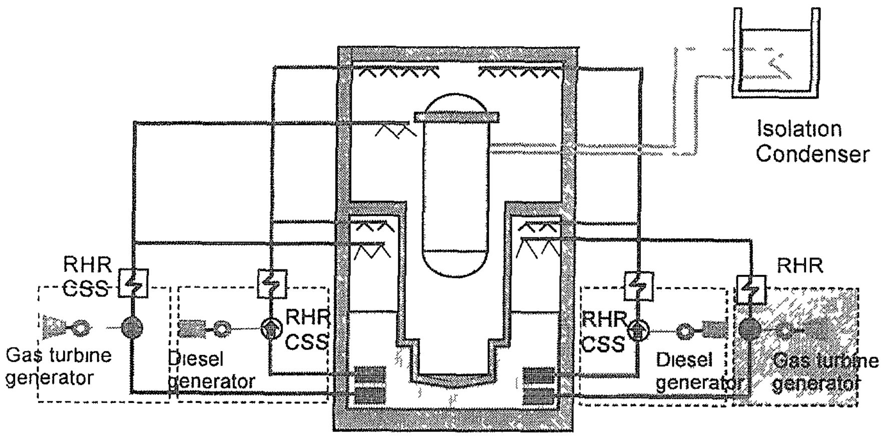

FIG. 8. Residual heat removal systems.

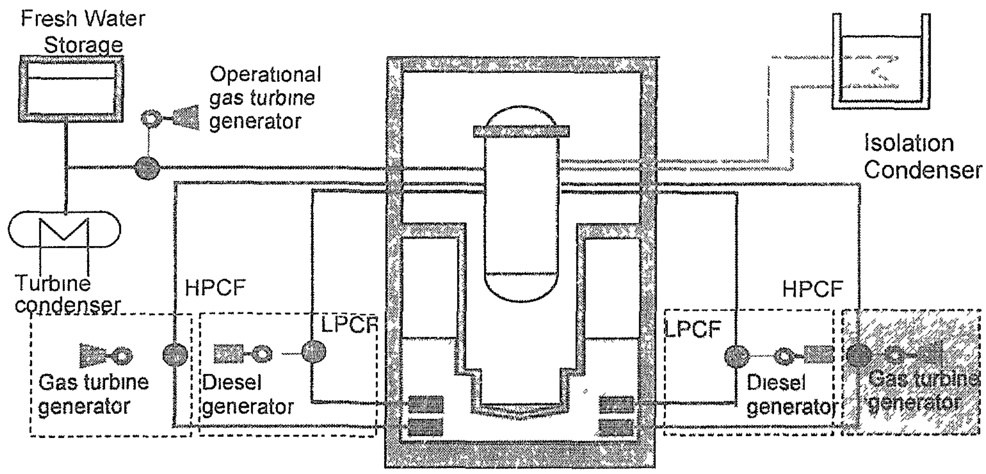

FIG 9. Core coolant make-up systems

Table 2. Comparison of main reactor data

		BWR 90	BWR 90+
Plant power output	MWe	1350	1500
Reactor power output	MWth	3800	4250
Cooling water temperature	°C	6	6
Number of recirculation pumps		8	12
Core coolant flow rate	kg/s	13900	17250
Fuel type		SVEA	SVEA Optima
Number of fuel assemblies		700	872
Average power per fuel assembly	MW	5,429	4,874
Volumetric power density	kW/l		54,9
Specific power density	kW/kgU$_{IN}$		28,5
Number of control rods		169	213
Number of scram groups		18	16
Number of safety/relief valves		18	16

BWR 75 design with independence and separation will in principle be kept. These principles have given cost savings in the field of operation and maintenance since they provide possibilities for inspection, testing and maintenance activities during normal operation. Probabilistic safety analyses, which are powerful tools for evaluating system configurations, show that the use of four identical components and subsystems has a certain disadvantage with respect to common-mode failures, however; a suitable diversification will yield favourable results with respect to the total reliability of the safety systems.

Therefore, a number of diversified components and system functions have been introduced, in line with the new safety requirements from the Finnish safety authority STUK, which demand diversity and provisions, in particular against loss of the final heat sink normally used, and that relief valves be actuated only temporarily. To this end, a passive heat removal system similar to the isolation condenser in the first ABB Atom NPP, Oskarshamn 1, will be incorporated in the design. The principles for the residual heat removal and coolant make-up systems are outlined in figures 8 and 9, respectively.

5.5 Comparison of main data

A brief summary of main reactor data for the BWR 90+ versus the BWR 90 is provided in Table 2.

6. CONCLUSIONS

ABB is convinced that there will be a reviving market for new nuclear power capacity in the near future. ABB is determined to be in a position to be able to compete for the new orders and therefore has continued its development efforts.

The BWR 90 design is available today for deployment in new plant projects and has been offered commercially. The design has, with a positive outcome, been subjected to a review by European utilities to evaluate how it compares with the requirements established by the EUR group.

The development of the BWR 90+ design is based on its 'forerunners' and can be referred to as a 'proven design', in line with power utilities' preferences. The 1500 MWe design incorporates considerations of new safety requirements, including severe accident impacts, and it will be marketed by the turn of the century.

The development work on the BWR 90+ design will also serve as input for modernisation, uprates and improvements of earlier generations of nuclear power plants.

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DESIGN CONCEPTS AND STATUS OF THE KOREAN NEXT GENERATION REACTOR (KNGR)

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Abstract

The national project to develop KNGR, a 4000 MWth evolutionary advanced light water reactor (ALWR), has been organized in three phases according to the development status in 1992. During the first phase, the top-tier design requirements and the design concepts to meet the requirements had been established. The project is currently in the second phase of which the major objective is to complete the basic design sufficient to confirm the plant safety. This paper describes the overall design concepts and status of the KNGR briefly which developed and/or being developed through the project.

1 INTRODUCTION

The KNGR is an evolutionary ALWR based on the current Korean Standard Nuclear Power Plant (KSNP) design with capacity evolution. It also incorporates a number of design modifications and improvements to meet the utility's needs for enhanced safety and economic goals and to address the new licensing issues such as mitigation of severe accidents.

The major evolution concept of the KNGR is based upon the results of research in the Phase I for two years. During this period, The design concepts had been set up to match domestic needs and capabilities through reviewing ALWR designs being developed by the advanced countries. To establish the safety and economic goals for the KNGR, the ALWRs were also compared quantitatively through safety and economic evaluation. The 42 top-tier design requirements had been established through this comparative study. The major requirements are as follows;

- General Requirements
  - Type and capacity: PWR, 4000 MWt (rated thermal power);
  - Plant lifetime: 60 years;
  - Seismic design: SSE 0.3g;
  - Containment building: Concrete double containment;
  - Safety goals: core damage frequency lower than 10^{-5}/RY and frequency of large radiation release due to containment failure lower than 10^{-6}/RY.
- Performance Requirements and Economic Goals
  - Plant availability: 90 %;
  - Occupational radiation exposure: less than 1 manSv/RY;
  - Construction period: 48 months; and
  - Economic goal: 20 % cost advantages over competitive energy sources.

As noticed, the KNGR aims at both enhanced safety and economic competitivity. From the point of view of probabilistic safety assessment (PSA), KNGR will have about 10 times lower probabilities of core damage and accidental radiation release than the KSNP. The economic goal of KNGR is considered achievable by high performance in operation and cost savings in construction.

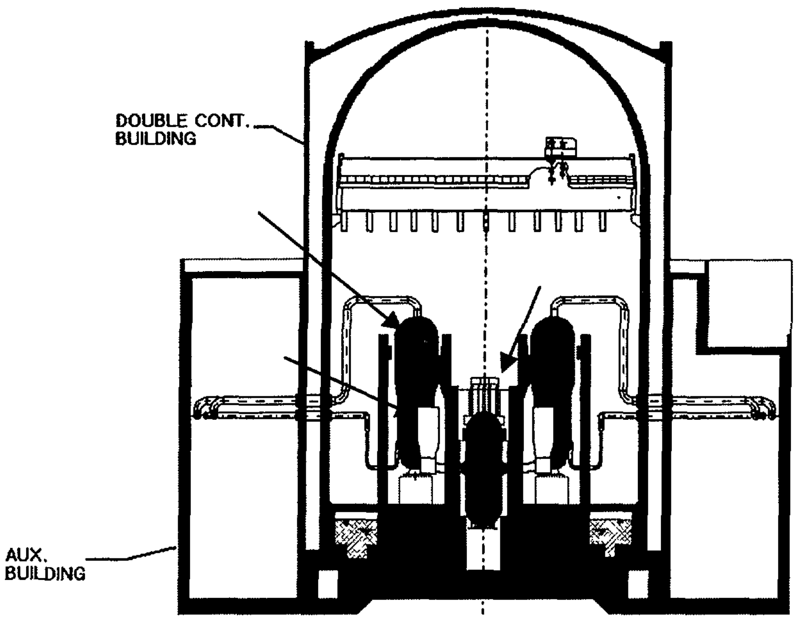


FIG. 1. A schematic diagram of arrangements of the primary components.

2 DESCRIPTION OF THE NUCLEAR SYSTEMS

2.1 Primary circuit and its main characteristics

The nuclear steam supply system of the KNGR is designed to operate at rated output of 4000 MWth to produce an electric power output of around 1350 MWe. The major components of the primary circuit are the reactor vessel, two coolant loops, each containing one hot leg, two cold legs, one steam generator (SG), and two reactor coolant pumps (RCPs), and one pressurizer (PZR) connected to one of the hot legs. Two SGs and four RCPs are arranged symmetrically. A schematic diagram of arrangements and locations of the primary components are shown in Figure 1.

The design temperature in the hot leg is reduced to lower than that of the currently operating plants in order to increase the operating margin and to reduce the SG tube corrosion problem.

The capacities of the PZR and the SGs (especially secondary side) are increased from that of current designs. The increased capacity of the pressurizer accomodates the plant transients without power operated relief valves. Conventional spring loaded safety valves mounted to the top of the PZR are replaced by the pilot operated safety relief valves (POSRVs), and functions of the RCS overpressure protection and safety depressurization could be performed by the POSRVs. The increased water inventory on the secondary side reduces the pontential for unplanned reactor trips and provide longer operator response time in case of the total loss of feed water accident.

2.2 Reactor core and fuel

The core consists of 241 fuel assemblies built up by fuel rods containing uranium dioxide fuel with an average enrichment of 2.6 w/o in a 16x16 array. Each fuel assembly consists of 236 fuel rods, 5 guide tubes. The number of CEAs is 93 with 8 spare CEAs. 17 of the 93 CEAs are part-strength CEAs. The absorber materials used for full-strength control rods are boron carbide (B_4C) pellets. Inconell alloy 625 is used as the absorber material for the part-strength control rods.

The core is designed for an operating cycle of 18-24 months with a discharge burnup as high as approximately 60,000 MWD/MTU, and has an increased thermal margin of up to 15 % to enhance safety and improve operation performance. A portion of the fuel rods contains uranium fuel mixed with a burnable absorber (Gadolinium) to suppress excess reactivity after fuelling and to help control the power distribution in the core. The neutron flux shape is monitored by fixed in-core instrumentation (ICI) assemblies and ex-core neutron flux monitoring system.

2.3 Primary component

The *reactor* comprises a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical closure head. The major design improvements incorporated in the reactor design include; larger operating margins, higher power level, and lower failure rate of fuel elements for higher plant availability and reliability. The lifetime of the reactor pressure vessel is improved to 60 years by use of low-carbon steel, which has lower contents of Cu, Ni, P, and S compared to current designs. The core support structures are designed to support and orient the reactor core fuel assemblies and control element assemblies, to direct the reactor coolant to the core. The core support barrel and the upper guide structure are supported at its upper flange from a ledge in the reactor vessel flange. The flange thickness is increased to sustain the enhaned seismic requirements.

In addition, KNGR adopted integrated head assembly (IHA) for availabilty improvement by reduction outage time. IHA is a mechanical assembly of various components required to provide lifting the reactor vessel closure head and its appurtenances, cooling of the CEDM, supporting the head area cables and protecting missiles generated from the reactor vessel head area. One piece removal of head assembly is estimated to save almost three days of refuelling outage time.

The *steam generators* are vertical U-tube heat exchangers with peerless type steam dryers, moisture separators, and an integral economizer in which heat is transferred from the reactor coolant to the main steam and feedwater system. A major improved feature incorporated into the steam generator design is the use of advanced corrosion resistant material in the steam generator tubes namely Inconel 690 replacing Inconel 600. In order to improve the operating margin of the steam generator, the tube-plugging margin is increased from 8 % in the ealier designs to 10%.

The *pressurizer* consisting of the pressurizer heaters, spray nozzles, and POSRVs maintains reactor coolant pressure and inventory in the reactor coolant system within specified limits. The pressurizer inventory is increased sufficiently so that RCS pressure can be maintained within the limit without the actuation of pressure relief devices in case of normal and upset conditions and the safety valve of pressurizer should not be opened under the transient conditions. The functional capability of the pressurizer is enhanced by using an increased volume relative to power and improved pressurizer heater control maneuverability during reactor shutdown.

The Leak-Before-Break (LBB) is adopted in the *piping* system of the KNGR, since the pipe whip restraint and the support of the jet impingement shield in the piping system of the earlier plant are expensive to build and maintain, and lead to a potential degradation of plant safety. The LBB technology is applicable to the main coolant lines, surge lines, and pipes of the shutdown cooling system, the safety injection system and the main steam line in the containment. This technology reduces the redundant supports of the pipe in the NSSS pipe system and thus, the construction cost of the plant.

The *reactor coolant pumps* circulate reactor coolant through the reactor vessel to the steam generators for heat removal and return it to the reactor vessel. The pump is a single-stage centrifugal unit of vertical type, which has a 12,000 horsepower motor. To assure leak-tightness of the shaft, the mechanical seal type is taken in order to seal against the full internal pressure in the pump and auxiliary seal injection pump is adopted to cope with station black out accident. The basic function and type of the pump in the current plant is just the same as the KSNP.

2.4 Operating characteristics

The power control system is capable of daily load follow operation with frequency control operation at a typical load variation profile in Korea; 16 hours at 100 % and 4 hours at 50 % with 2 hours ramps for power decreases and increases. To do this, Mode K algorithms, which will control core power automatically according to power demand, are under development.

The load rejection capability at the rated power should also be incorporated. This capability can reduce the outage time caused by the secondary system troubles since the reactor power can be brought up to 100% as soon as the troubles have been fixed.

3 DESCRIPTION OF TURBINE GENERATOR PLANT SYSTEM

The turbine generator plant consists of main steam, extraction steam, feedwater, condensate, turbine generator and auxiliary systems. For these systems, heat balance optimization studies have been made, considering system operability, reliability, availability and economy. The turbine generator systems are designed to be capable of operation at 3 % house load for a period of at least 4 hours without any detrimental effects of the systems, and capable of startup to full load from cold conditions in 8 hours, including rotor preheat. The main steam lines and the high pressure turbine are designed for a steam pressure of 6.9 MPa (1,000 psia), and two reheater stages are provided between the high pressure and the low pressure turbines. The generator is a three-phase, 4-pole unit operating at 1800 rpm.

The feedwater pump configuration is selected to be 3x50 % because of its ability to allow more reliable operation; all three pumps are normally operating, and the plant can remain at 100% power operation if one of the feedwater pumps is lost. On-line condensate polishers, that can operate in full and partial flow, as well as in bypass mode, are provided to maintain proper water chemistry during normal power operation. In the feedwater systems, the feedwater heaters are installed in 7 stages and arranged horizontally for easy maintenance and reliability.

4 INSTRUMENTATION AND CONTROL (I&C)

KNGR is equipped with a digitized Man-Machine Interface System (MMIS), which encompasses the control room systems and Instrumentation and Control (I&C) systems, reflecting the modern computer technology.

One of the main features of the I&C system is the use of microprocessor-based multi-loop controllers for the safety, including reactor protection and non-safety control systems. Engineering workstation computers and industrial personal computers are used for the two diverse data processing systems, respectively. To maintain the plant safety against common mode failures in software (S/W) due to the use of digital systems, controllers of diverse types and manufacturers will be employed in the control and protection systems. For data communication, a high-speed fibre-optic network is used. A remote signal multiplexer is also utilized for the safety and non-safety systems field signal transmission to save considerable amount of cables and cable trays. Since the S/W is heavily relied upon in fully digitized MMIS, a stringent S/W qualification process will be established and followed for the life cycle of the KNGR. The MMIS concept to be implemented in the KNGR design is schematically depicted in Figure 2.

The KNGR MCR design is characterized by 1) redundant compact workstations for operators; 2) seismically qualified Large Display Panel (LDP) for overall process monitoring of the plant to be shared among operating crew; 3) multi-functional soft controls for discrete and modulation control; 4) computerized procedure system to provide on one of the workstation CRTs with context sensitive operation guides, operational information, and navigation links to the soft controls for normal and emergency circumstances; and 5) safety console for dedicated conventional miniature button type

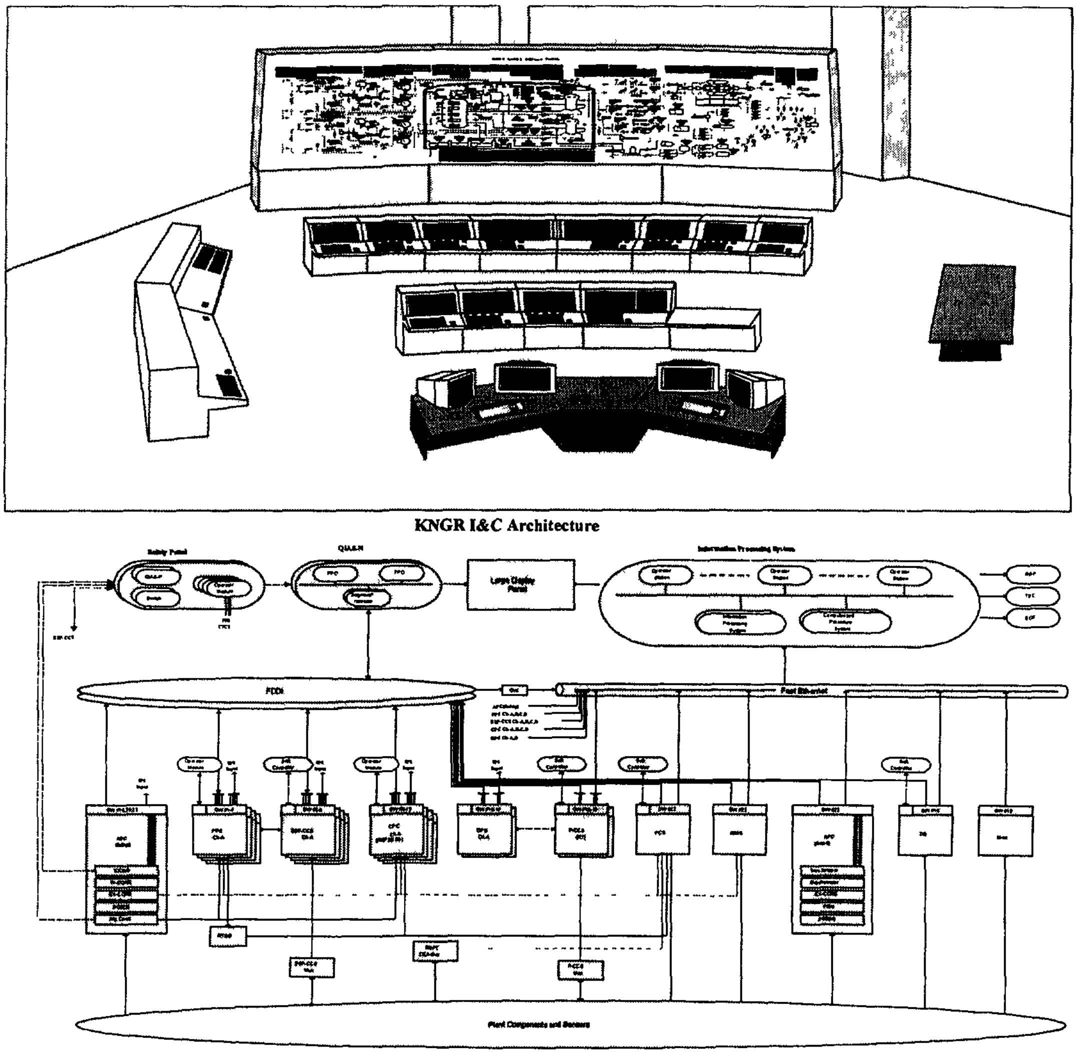


FIG. 2. KNGR Man-Machine Interface System (MMIS).

controls provided to control essential safety functions. CRTs and FPDs (flat panel displays) are extensively used for presentation of operational information.

The human factor engineering is an essential element of the control room facility design and Man-Machine Interface (MMI) design and its principles are systematically employed to ensure safe and convenient operation. Operating experience review analysis, function analysis, and task analysis are performed to provide systematic input to the MMI design.

Partial dynamic mockup has been constructed based on the simulator of predecessor plant (KSNP) system models. This facility is used to perform initial verification of suitability of the MMI design. In the forthcoming design stage, the mockup will be expaned for intermediate validation of the design and I&C prototyping will be undertaken for smooth development of KNGR MMIS facility.

5 ELECTRICAL SYSTEMS

The main power system consists of the generator, the generator circuit breaker, the main transformer, the unit auxiliary transformer and the stand-by transformer. The normal power source for non-safety and permanent non-safety loads is the off-site power source and the generator. If the normal power source is not available, the permanent non-safety loads are covered by two alternative sources: one from the stand-by off-site power source (via the stand-by transformer) and the other from one non-1E alternate AC power source with a gas turbine generator.

The electric power necessary for the safety-related systems is supplied through 4 alternative ways:

1. the normal power source, i.e., the normal off-site power and the in-house generation,
2. the stand-by off-site power, i.e., the off-site power connected through the stand-by transformer,
3. the on-site standby power supply, i.e., two diesel generators, and
4. the alternative AC source, i.e., the gas turbine generator.

The on-site power supply is ensured by two independent Class 1E diesel generator sets; each of them is located in a separated building and is connected to one 4.16 kV safety bus. The alternate AC source adds more redundancy to the electric power supply even though it is not a safety grade system. The non-class 1E alternate AC is provided to cope with Station Blackout (SBO) situation which have a high potential of transients progressing to severe accidents. The alternate AC source is sized with sufficient capacity to accommodate the loads on the safety and the permanent non-safety buses.

6 SAFETY CONCEPT

6.1 Safety goals and design philosophy

One of the KNGR development policies is to increase the level of safety dramatically. To implement this policy, the plant will be designed in accordance with the established licensing design basis to meet the licensing criteria and also be designed with an additional safety margin in order to improve the protection of the investment, as well as the protection of the public health.

The safety goals of the KNGR can be summarized as follows;

- The total core damage frequency should not exceed 10E-5 per year, considering both internal and external initiating events.
- The whole body dose for a person at the site boundary should not exceed 0.01 Sv (1 rem) during 24 hours after initiation of core damage even in the event of containment failure. The probability exceeding such a limit should be less than 10E-6 per year.
- The frequency of an accident in which the release of long-lived radioisotopes such as Cs-137 would exceed the amount to limit the land use shall be less than 10E-6 per year.

In addition to the public safety, a concept of investment protection will be implemented in the KNGR design. In KNGR, there are many investment protection goals such as loss-of-coolant-accident (LOCA) protection, steam generator inventory, and so on. For example, following the event of a small break LOCA up to 15 cm pipe break, continued use of the reactor with its fuel should be possible.

Another important design philosophy for safety is the increased design margins. A few examples of the design requirements following this philosophy are the requested core thermal margin of 10~15 %, sufficient system capacity for operator recovery action time of more than 30 minutes, and station blackout coping time of 8 hours.

6.2 Safety systems and features

6.2.1 Active Safety Systems

The active safety systems consist of the safety injection system (SIS), safety depressurization system (SDS), in-containment refuelling water storage system (IRWST), auxiliary feedwater system (AFS), and containment spray system (CSS).

The main design concept of the SIS is simplification and redundancy to achieve higher reliability and better performance. The safety injection lines are mechanically 4 trains and electrically

2 divisions without the tie branch between the injection lines for simplicity and independence. Each train has one safety injection pump and one safety injection tank. The common header currently installed in the SIS trains is eliminated and, finally, functions for safety injection and shutdown cooling are separated. Through the IRWST the current operation modes of high pressure, low pressure, and re-circulation can be merged into only one operation mode (i.e., safety injection). The emergency cooling water is designed to be injected directly into the reactor vessel so that the possibility of spill of the injected flow through the broken cold leg is eliminated.

The refuelling water storage tank is located at the inside of the containment and the arrangement is made in such a way that the injected emergency cooling water can return to the IRWST. The susceptibility of the current refuelling water storage tank to external hazard is lowered by locating it at the inside of the containment. The functions of IRWST are as follows; the storage of refuelling water, a single source of water for the safety injection, shutdown cooling, and containment spray pumps, a heat sink to condensing steam discharged from the pressurizer for rapid depressurization if necessary to prevent high pressure core melt or to enable feed and bleed operation, and coolant supply to the cavity flooding system in case of severe accidents to protect core melt.

The AFWS is designed to supply feedwater to the SGs for RCS heat removal in case of loss of main/startup feedwater systems. In addition, the AFWS refills the SGs following a LOCA to minimize leakage through pre-existing tube leaks. The AFWS is a 2 divisions and 4 trains system. The reliability of the AFWS has been increased by use of two 100 % motor-driven pumps, two 100 % turbine-driven pumps and two independent safety-related emergency feedwater storage tanks as a water source instead of condensate storage tank.

6.2.2 Passive safety systems

KNGR is an evolutionary plant that relies on the active systems for ensuring its safety. However, some passive systems are under research to be incorporated in the design. The three main passive safety systems under development are:

- A fluidic device that is located at the discharge of the safety injection tank (SIT). It is a passive system to inject the borated water into the RCS in a passively regulating way. The system has a capability to reduce the discharge flow to 10 % of the maximum flow. This system is expected to enhance the performance against the loss of coolant accidents by lengthening the water injection time.

- The Passive Secondary Condensing System (PSCS) which secures the passive heat removal through the steam generators in case of the loss of AFWS. The PSCS locates outside containment. The PSCS takes inlet flow into the isolation condenser submerged in the condenser tank from the steam line and discharges outlet flow into the feedwater line. This connection can make it possible passive core cooling without any active component running by natural circulation.

- Fusible plugs in the reactor cavity. The reactor cavity and the IRWST are directly connected with each other. The IRWST side is open for coolant supply, while the reactor cavity side is normally closed by end plugs that are made of fusible metal. If, in the event of an accident, the ambient temperature of the reactor cavity increases to the melting temperature of the fusible metal, the plugs would start melting to supply cooling water from the IRWST to the reactor cavity.

6.3 Severe accidents (Beyond design basis accidents)

The measures to cope with severe accident are divided into two categories, prevention and mitigation.

Severe accident prevention features can be summarized as follows:

- increased design margin such as larger pressurizer, larger steam generators, and increased thermal margin,
- reliable ESF systems such as SIS, AFWS, and CSS,
- extended ESF systems such as SDS with IRWST, AAC, and PSCS, and
- containment bypass prevention.

Severe accident mitigation features and strategy can be summarized as follows;

- hydrogen mitigation system such as passive auto-recombiner and hydrogen ignitor,
- wide reactor cavity area and cavity cooling system,
- SDS and IRWST,
- robust double containment with large volume, and
- in-vessel retention of molten core (the feasibility study is in progress).

7 PLANT LAYOUT

7.1 Buildings and structures, including plot plan

The general arrangement of KNGR has been developed based on the twin-unit concept and slide-along arrangement with common facilities such as the radwaste building as shown in Figure 3(a). The auxiliary building is designed to surround the containment building to achieve 4-train design concept for the safety systems. This type of auxiliary building prevents the direct access of the containment during construction. To solve the problem and shorten the construction schedule more, the applicability of the over-the-top method using the large cranes for NSSS main equipment installation in the containment has been reviewed. Another method we are reviewing for the reduction of the construction period is the deck plate method for the auxiliary building floor. The result of the present analysis indicates that the use of the deck plate method reduces the construction period by two months.

The auxiliary and containment buildings will be built on a common basemat. The common basemat will improve the resistance against seismic events and reduce the number of walls between buildings so that rebar and formwork cost can be reduced.

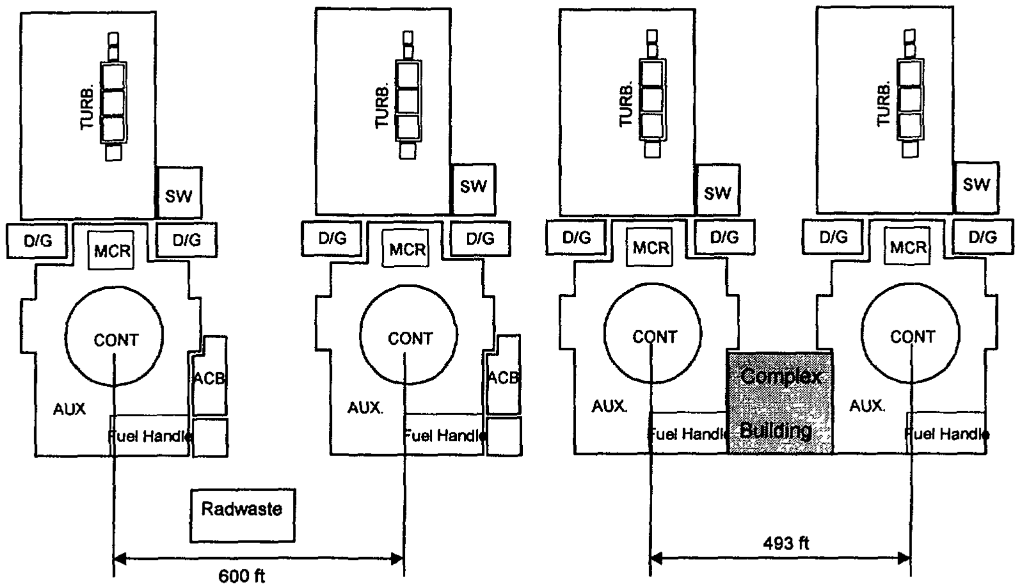
The KNGR plant consists of the nuclear island which has containment building, auxiliary building, diesel generator building, access control building, and radwaste building and turbine island which has turbine building and annex building.

In addition to the current arrangement, KNGR has been considered to have a complex building, which consists of access control building, radwaste building, and hot machine shop as shown in Figure 3(b). The purpose of this building, which is located at the center of two plant, is to enhance operability by the effective facility management.

7.2 Containment building

The containment building consists of a reinforced concrete outer containment, a steel-lined, post-tensioned concrete inner containment, and a reinforced concrete internal structure. The containment building houses the pressurized water reactor, steam generator, reactor coolant loops, In-Containment Refuelling Water Storage Tank (IRWST), and portions of the auxiliary systems.

The containment building is designed to provide biological shielding, external missile protection, and to sustain all internal and external loading conditions that may reasonably be expected to occur during the life of the plant. The equipment hatch, which has 7,8 m (26 feet) inner diameter, is selected to accommodate one-piece replacement of a steam generator. A polar bridge crane is



(a) Standard general arrangement (b) general arrangement with complex building

FIG. 3 The General arrangement with complex building of KNGR.

supported from the containment wall. The bridge crane has the capability to install and remove the steam generators.

The outer containment is composed of a reinforced concrete right cylinder with a shallow, domed roof. It has an inner radius of 25,8 m (86 feet). Annular space, called the Annulus, is provided between the inner radius of the Outer Containment and the outer radius of the inner containment above the basemat. The main function of the Annulus is for collection of post-LOCA containment atmosphere leakage. This leakage is filtered, recirculated, and released by the annulus ventilation system. Adequate access is provided for installing, testing, inspecting, and tensioning the tendons.

The inner containment is a post-tensioned concrete cylinder with a hemispherical dome. There is no structural connection between the free-standing portion of the inner containment and the adjacent structures other than penetrations and their supports. The lateral loads due to seismic and other forces are transferred to the foundation concrete through the structural concrete reinforcing connections. The containment free volume has been set at $9{,}1 \times 10^4$ m^3 (3.2×10^6 ft^3).

7.3 Turbine building

The turbine building houses the turbine generator, the condenser systems, the preheater system, the condensate and feedwater systems, and other systems associated with power generation. The turbine building is classified as non-safety related. It has no major structural interface with other buildings except for a seismic interface with the connecting auxiliary building. It is designed such that under SSE conditions, its failure will not cause failure of safety related structures. The turbine building is located such that the containment building is on the projection of the turbine shaft, on the high pressure turbine side. This allows for optimization of the piping and cable routes to the Nuclear Island. This arrangement also minimizes the risk of damage to safety-related equipment by missiles from the turbine or the generator, in the event of an accident.

7.4 Other buildings

To assure the safety and reliability, the auxiliary building is designed to enhance physical separation for mitigation of internal flooding, fire propagation as well as security and sabotage. The auxiliary building shares with fuel building in a quadrant arrangement. The auxiliary building houses pumps and heat exchangers for safety injection system and safety cooling system. Also, the emergency feedwater tanks and main control room are located in the auxiliary building

The radwaste building designed to be shared between two units houses the liquid waste, gaseous waste, and solid waste systems. In accordance with Reg Guide 1.143, it is designed to provide protection against natural phenomena and to accommodate associated environmental conditions to the extent necessary to retain the spillage of potentially contaminated solids or liquids within the building. It has no major structural interface with other buildings.

The Emergency Diesel Generator (EDG) buildings are located at either side of the auxiliary building. Each EDG supports one division. These buildings are seismic category I structures which provide protection from fire, missiles, and the environment The EDGs are arranged as separate entities with dedicated auxiliaries including air supply, exhausts, and cooling systems, so that they are independent of one another in all respects. The EDG buildings are arranged to provide routine maintenance facilities and maintenance access space such that work on one EDG in no way affects the operability of the other EDG.

8 PROJECT STATUS AND SCHEDULE

The KNGR development project was organized in three phases related to the development status, and the third phase ends with completion of a detailed standard design. Phase I of the project was scheduled to run for the two-year period from the end of 1992 to the end of 1994, and the major activity was to develop top tier design requirements and concepts for the new design. Phase I was finished according to the plans, and is now followed by Phase II which is a four-year programme, running from 1995 to 1999. The major activities of this phase are to develop a basic design for a licensing review, to ensure the safety of the KNGR and thus, its licensibility. The level of design completion by the end of the Phase II is estimated to be around 20% of the total engineering works needed for construction, and commissioning, of a plant.

The major activities in the Phase II programme that are currently under way, can be categorized as a basic study, development of detailed user's requirements, regulatory research, development of an information management system, and design activities. Licensing interactions with the regulatory body becomes also an important part as the design work proceeds. In addition, some supporting research and development have been conducted to upgrade the in-house engineering capability and investigate design alternatives. Also, we are testing and preparing to test some features, such as the Direct Vessel Injection System, Fluidic Device, and so on, of the newly adopted advanced and passive design features to KNGR to verify thier functions and designs.

The plant level analyses of safety, economics, availability, and constructibility have been performed periodically during the Phase II programme to achieve the safety and economic goals for KNGR through the feedback of the analyses results. The current analysis results show that KNGR meets all safety and economic goals except availability goal. In order to increase the availability, some design improvements such as integrated head assembly, permanent pool seal, and blind flange are considered.

The basic design and safety analysis will be finished end of 1998. After that, the standard safety analysis report will be prepared for licensing review to get the design certification and the design specification for long lead items such as reactor vessel, steam generators, and reactor coolant pipings will be issued until Feb. 1999.

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Abstract

Nuclear power generated by light water reactors accounts for approximately 1/3 of Japan's power supply. Development of the Advanced Pressurized Water Reactor (APWR) was initiated by five PWR electric power companies (Hokkaido, Kansai, Shikoku, Kyushu and Japan Atomic Power), Mitsubishi Heavy Industries, and Westinghouse, with a view to providing a nuclear power source to meet future energy demand in Japan. The APWR was developed based on the results of the Improvement and Standardization Program, promoted by the Ministry of International Trade and Industry, with reconsideration of the needs of age, such as construction cost reduction, enhanced safety and increased reliability. One of the important concepts of the APWR is its large power rating that decreases the construction cost per unit of electric generation capacity. Though the electric output was lower at the early stage of basic design than it is now, uprating to approximately 1530 MW is achieved based on the results of design progress and high efficiency improvements to the steam turbine and reactor coolant pumps. Furthermore, the APWR remarkably enhances reliability, safety operability and maintainability by introducing new technologies that include a radial reflector and advanced accumulators. The first APWR is planned to be built at Tsuruga No. 3 and No. 4 by the Japan Atomic Power Company and will be the largest commercial operation plant in the early 21st century.

1. INTRODUCTION

The advanced PWR has been developed as a nuclear power plant for future use in Japan, in a joint international co-operative development project by seven companies comprising the five PWR electric power companies (Hokkaido, Kansai, Shikoku, Kyushu and the Japan Atomic Power company), Mitsubishi Heavy Industries and Westinghouse. Its development was part of the Improvement and Standardization Program of the Ministry of International Trade and Industry. In the APWR, advanced technologies based on the operational experience gained up to now have been incorporated. Furthermore, the performance, ease of control, reliability, and safety of the plant have been increased, and the construction cost reduced, due to the benefits of scale resulting from the increase in capacity. Here we introduce some outstanding features of this new APWR.

2. DESCRIPTION OF NUCLEAR SYSTEMS

2.1 Primary circuit and its main characteristics

Table 1 shows a comparison of some major parameters between the APWR and a current four-loop plant. The APWR is in the largest capacity class of LWRs in Japan and has adopted high performance steam generators and a highly efficient turbine with 54 in. (1370 mm) last stage blades. Various improvements have been incorporated in the reactor core in order to reduce uranium requirements, and to provide increased flexibility for various applications such as the use of MOX cores and high burn-up fuels.

Critical equipment such as core internals and steam generators have been designed with the operational experience of aging at existing plants taken into account so that a high degree of reliability can be obtained. To ensure safety, the reliability of the equipment and systems has been increased. For instrumentation and control systems, the latest digital control technologies have been incorporated not only in the reactor control system but also in the reactor protection system. As well, the latest electronics technologies to improve the man machine interface have been introduced in the main control room. In order to make the plant easier to maintain a variety of improved technologies have been incorporated, thus improving the efficiency of periodical inspections and reducing exposure for employees.

TABLE 1 MAJOR APWR PARAMETERS

	Current four-loop PWR	APWR
Electric power output	1,180 MWe	Approx. 1,530 MWe
Thermal power	3,411 MWt	Approx. 4,450 MWt
Fuel type	17 × 17	Improved type 17 × 17
Number of fuel assemblies	193	257
Fuel effective length	Approx. 3.7 m	Approx. 3.7 m
Total uranium inventory	Approx. 89 MTU	Approx. 120 MTU
Reactor vessel	Approx. 4.4 m inner dia. & 13 m height	Approx. 5.2 m inner dia. & 14 m height
Steam generator Heat transfer surface area (m^2)	4870	6500
Primary system flow ($m^3/h/loop$)	Approx. 2.01×10^4	Approx. 2.58×10^4
Turbine	TC6F44	TC6F54
Containment	PCCV	PCCV
Engineered safety systems	Two trains	Four trains of mechanical systems
Refueling water storage	Outside containment	Inside containment
Reactor protection system	Analog	Digital
Reactor control system	Digital	Digital
Main control room	Standard	Improved

2.2 Reactor core and fuel design

The reactor core, consisting of 257 improved 17 x 17 fuel assemblies, has a thermal output of approximately 4450 MWt. Considering the need to reduce fuel cycle costs and future needs for MOX reactor cores and high burn-up cores, a variety of improvements have been incorporated in the reactor core. The core has also been designed so that it can use plutonium fuel with 1/3 or more MOX cores, and is flexible enough to use fuel with a burn-up having long operating cycles.

In order to reduce fuel cycle costs, the fuel assemblies are provided with Zircaloy grids to reduce neutron absorption; the core is surrounded by a reflector to reduce neutron leakage, thus increasing neutron efficiency.

The reactor uses improved 17 x 17 fuel assemblies based on the 17 x 17 fuel that has operated well in existing plants. The design adopts Zircaloy grids, as described above, and longer plenum in the fuel rod, allowing it to be used for high burn-ups and increased loadings of MOX fuel.

2.3 Primary system components

2.3.1 Reactor pressure vessel

Although the reactor vessel inside diameter has been increased to approximately 5.2 m in order to accommodate 257 fuel assemblies, the vessel is made with forged rings and has no longitudinal welds in the core area, in contrast to the latest four-loop plants. The neutron irradiation of the steel opposite the core has been reduced to approximately 1/3 of present reactors by installing a radial reflector, thus increasing the reliability of the reactor vessel.

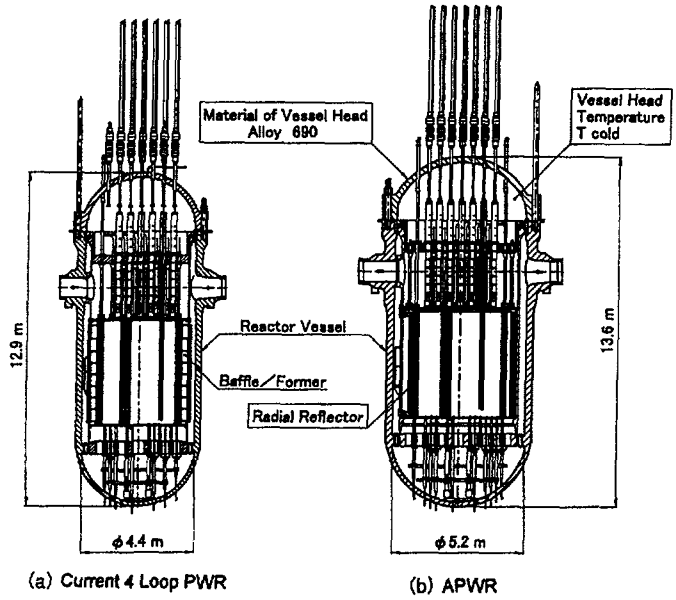

FIG. 1. A comparison between the APWR and a current PWR.

In order to reduce susceptibility to corrosion cracking of the penetrations of the reactor vessel head, the nozzle material has been improved (from alloy 600 to heat-treated alloy 690) and the primary coolant temperature in the reactor vessel head is designed so that it is reduced to same temperature as the reactor inlet.

A comparison between the APWR and a current four-loop plant is shown in Figure 1.

2.3.2 Reactor internals

The radial reflector, consisting of eight rings of stainless steel blocks, not only reduces fuel cycle costs but also reduces the irradiation of the reactor vessel and core internals. Through installation of the radial reflector, neutron irradiation of the reactor vessel can be reduced to 1/3 that of present reactors. In present reactors, the core baffle is a plate structure held together with 2000 or more bolts, whereas the new radial reflector has a simple construction which does not use bolts in the core region. Moreover, there are no welds in the core region. As a result, a high reliability can be

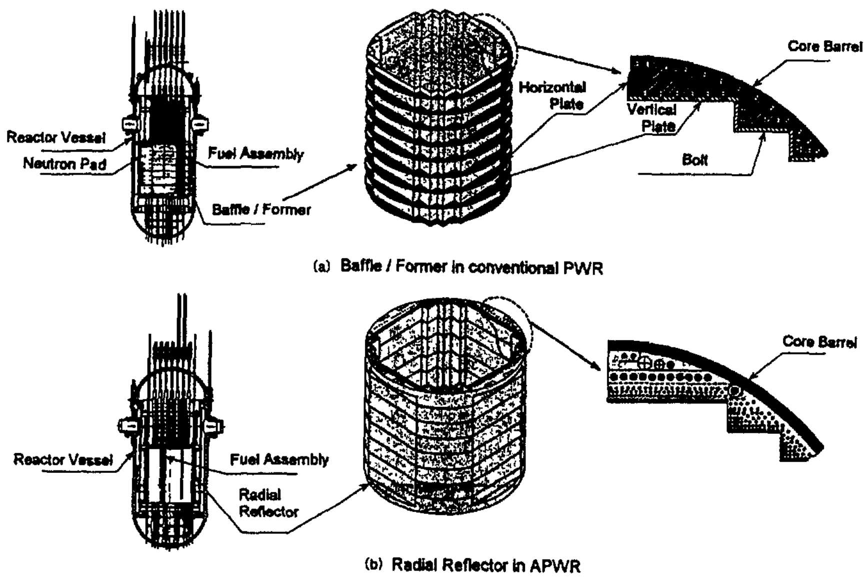

FIG. 2. APWR radial reflector.

expected, even for a longer design life. Figure 2 shows the radial reflector compared with the conventional PWR structure.

In addition to the radial reflector, other components have been improved to reduce vibration due to fluid flow, thus reducing wear of the lower support plate that stabilizes the flow inside the lower plenum.

2.3.3 Steam generators

The APWR has adopted large type steam generators (Type 70F-1) having an increased capacity to match the increased capacity of the reactor core. The heat-transfer tubes are 3/4 inch (19 mm) in diameter, which is smaller than the 7/8 inch (22 mm) used in existing plants. This reduced size results in a more compact steam generator that is more resistant to earthquakes.

The heat-transfer tubes of the steam generators are made of thermal treated alloy 690 (TT690). The design of the anti-vibration bar in the U-bend area of the heat-transfer tubes has been improved to reduce the risk of flow induced vibration of the heat-transfer tubes.

In addition, in order to make it easier to do maintenance and inspections, accessibility has been improved by increasing the diameter of the manholes and in other ways. Figure 3 shows several design features of the APWR steam generator.

2.3.4 Reactor coolant pumps

Because the primary coolant pump has to supply a flow approximately 30% larger than that of an existing primary coolant pump, a 100A type pump (60 Hz), which is larger in capacity than the existing 93A-1 type, has been adopted. Heat-resistant 0-rings, as well as ceramic material that has larger size and excellent durability, are employed for the improved No. 1 seals, aiming at enhancement of the reliability during both normal and abnormal operation.

2.3.5 Main coolant pipe

For piping material, low alloy steel with stainless steel lining has been chosen from the point of view of enhanced reliability and inspectability.

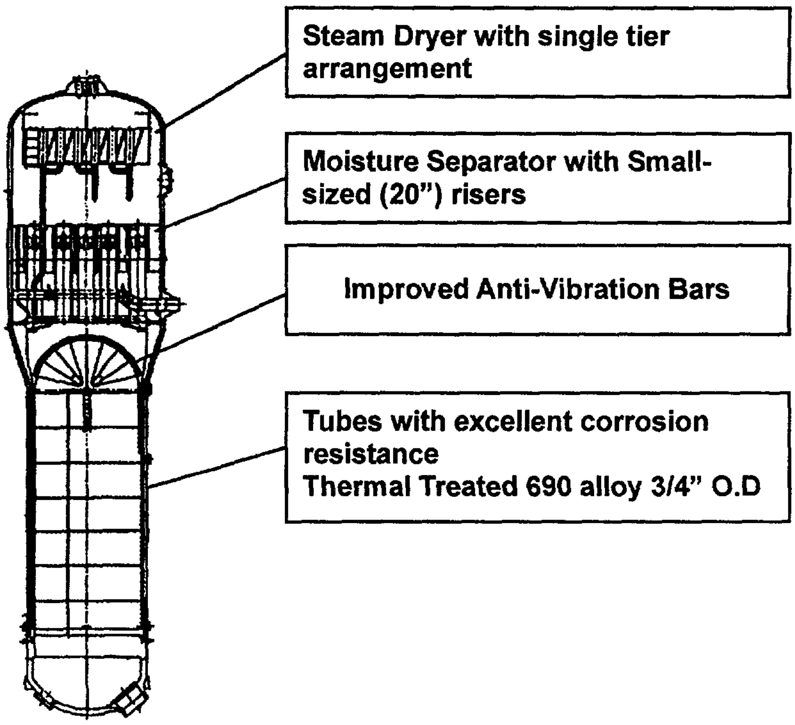

FIG. 3. APWR steam generator.

2.4 Reactor auxiliary systems

2.4.1 Reactivity control system

The reactivity control system injects highly borated water via the direct vessel injection (DVI) nozzle to the core, using two emergency boration pumps that take water from the boric acid tank (commonly used by chemical and volume control systems).

The major purpose of this equipment installation is to inject boron into the reactor following an excessive cooling event, such as a main steam pipe break, in order to rapidly achieve sub-criticality of the core after the occurrence of the accident.

2.5 Operating characteristics

The reactor is designed so that it can be operated automatically within the range of 15 to 100% of rated power by the reactor control system. Even in the low power range below 15%, the control rod control system can control the reactor automatically in the low power operating mode.

Usually the primary coolant average temperature is controlled to a programmed value that increases linearly with turbine output. When daily load following is used, however, the primary coolant average temperature is controlled to a variable reference temperature in order to reduce the amount of water that has to be processed.

The reactor control system is designed so that it can follow the load changes listed below without causing a reactor trip.

- a 10% step load change (within the range of 15 to 100%)
- a 5% per min ramp load change (within the range of 15 to 100%)
- a 100% load rejection

With respect to the load fluctuation to meet the grid demand, the following is planned:

- Daily load-follow operation of 100%-50%-100%
- Automatic frequency control or governor control to control system frequency up to 5% power.

3. DESCRIPTION OF TURBINE GENERATOR PLANT SYSTEM

3.1 Turbine generator system

The high-pressure turbine is a double flow turbine in which reaction blading is used in all stages.

Each low-pressure turbine is a double flow turbine with reaction blading. The last stage blades are 54 inch (1370 mm) blades that increase the electric power output and efficiency. The high performance blades, multiple seals, and the new high performance guide vanes further increase the efficiency. In addition, the moisture extraction system has been improved to reduce erosion.

The 54 inch last stage blade has been subjected to vibration tests and actual load tests in order to demonstrate that the turbine will have high performance and reliability. The performance has also been demonstrated in actual machines.

The moisture separator/reheater has a two-stage heater, and can achieve high efficiency. The turbine building has been reduced in size by reducing the outside dimensions of the moisture separator and adopting the so-called four-neck heater system, in which four feed heaters are installed in the neck of the condenser, in contrast to a conventional plant, which usually has two low-pressure feed heaters in this location.

The approximately 1530 MWe electric generator, which is of 4-pole type, has a larger rotor diameter than previous machines in order to increase the output. The rotor windings are cooled by hydrogen gas, while water is used for cooling of the stator. Excitation is provided by a brushless static system.

3.2 Condensate and feedwater systems

The feed heaters in the condensate and feedwater systems use seven extraction stages in order to increase the efficiency. In particular, the deaerator is installed on the upstream side of the final high-pressure feed heater to obtain the benefits of a direct contact heat exchanger.

The condensate and feedwater systems consist of 3 trains of five stages of LP heaters, the deaerator and 2 trains of single stage HP heaters.

The feed heater tubing material is expected to be changed to stainless steel, in order to avoid corrosion of the LP feed heater tubes caused by the ammonia added for control of water purity.

The capacity of the pumps installed in the condensate system is 2 x 50% units, plus one unit as spare. The capacity of the pumps installed in the feedwater system is 2 x 50% units with one 25% spare unit.

4. INSTRUMENTATION AND CONTROL SYSTEMS

4.1 Design concepts including control room

The main control room is provided with compact consoles on which CRTs and flat display panels are mounted. Conventional operating and monitoring devices such as switches, lamps, indicators, and recorders have been eliminated.

The plant is operated by touch-screen operations, and the monitoring information necessary for operation is displayed on the same screens as those used for operating the plant. Therefore, the work load of the operators is reduced and the reliability of operation is increased.

On the wall of the main control room, a large display panel is installed to display the major monitoring parameters for normal and abnormal conditions of the entire plant. Thus, the current status of the entire plant can be understood by everyone and communication between operators is improved.

4.2 Reactor protection system and other safety systems

The reactor protection system and other safety systems are digital and functionally distributed.

The reactor protection system consists of four channels, including the reactor trip breakers. Each channel is formed with multiple digital devices so as to provide redundant protection functions and to separate the reactor protection function from the other safety system operating functions.

The other safety systems consist of two trains. Each train has two sets of digital devices. To interface these systems with the auxiliary equipment in the plant, remote input/output devices arranged in a distributed fashion throughout the plant are connected to the host computer through optical fiber cables, thus reducing the volume of wiring.

The reactor protection system and other safety systems are provided with automatic test equipment so that periodical tests can be carried out fully automatically.

To achieve high reliability, the software used for the digital safety systems is modularized and simplified, with symbolic language used in the design. Verification and validation tests are to be carried out to the maximum extent possible.

In addition, as a countermeasure against accidental problems with the safety system software resulting in any common cause failure, we plan to install backup switches in the main control room for those critical safety functions for which there is sufficient time for operator action.

5. ELECTRICAL SYSTEMS

5.1 Operational power supply systems

The operational AC power supply system can receive external power from the main power supply system and from the standby power supply system. Power from the main power supply system comes through the main transformer and plant transformers. When the plant is operating normally, the main generator is connected to the external power system, and when the main generator is stopped, it is disconnected from the external power system by the generator load break switch GLBS. Therefore, the operational AC power supply system can receive power through the plant transformers continuously independent of whether the main generator is operating or not. If the main power supply system fails and the plant does not continue to operate independently, power will be received through the standby power supply system. Power from the standby power supply system is received through the standby transformer, which has sufficient capacity to maintain the plant, or hot standby conditions, and enable it to be shut down safety.

The buses of the operational AC power supply system are divided into two main groups:

- 6.6 kV high voltage and 440 V low voltage systems, each comprising normal buses to supply power to loads such as primary coolant pumps, feed pumps and other equipment required for normal plant operation;
- two trains of emergency buses to supply power to loads such as high-pressure injection pumps and other equipment required for the safety of the plant.

In addition to the above AC power supply systems, an AC power system has been provided, which can be supplied from batteries in the event of an interruption or total loss of all AC power, as has instrumentation and control power supply (consisting mainly of inverters) for supplying power to the instrumentation and control equipment that are mainly computer loads.

5.2 Safety-related systems

The emergency power systems for supplying power to the operating power systems when an accident occurs in the plant, or there is a loss of external power, include the emergency diesel generators and battery equipment. The emergency diesel generators start automatically and immediately, if an accident occurs or external power is lost, to supply power to the emergency buses. The emergency power systems are configured redundantly, and the safety of the plant can be secured with only one train of emergency power.

The DC power supply system can supply power to the instrumentation control power system during an instantaneous power failure. It has sufficient capacity to also supply the switchgear that must operate following a loss of external power, as well as to supply the excitation current for the diesel generators. Furthermore, it has sufficient capacity for maintaining the safety of the plant following a total failure of all AC power.

The bus configuration for the emergency power system is designed so that it is consistent with the configuration of the plant safety systems. As a result, the AC and DC power systems are divided into two trains, and the instrumentation and control power system is divided into four trains to be consistent with the four channels.

6. SAFETY CONCEPTS

6.1 Safety requirements and design philosophy

The configuration of the ECCS mechanical systems has been changed from the conventional two trains to four trains to give more redundancy and independence. Tie lines between loops have been eliminated to simplify the systems and increase the reliability.

In existing plants, refilling of the reactor vessel and re-flooding of the reactor core after a LOCA accomplished by both the accumulators and low-pressure injection pumps. In the APWR, however, advanced accumulators with two-stage discharge characteristics have been adopted and the conventional low-pressure injection pumps have been eliminated to simplify the equipment and increase reliability. The refuelling water pit is installed inside the containment, thus eliminating the operation of changing the suction from the refuelling water tank to the containment recirculation sump, which is necessary during an accident in existing plants. A comparison between the ECCS of an existing plant and the APWR is shown in Figure 4. The APWR is expected to have a core damage frequency that is at least one order of magnitude lower than that of an existing plant.

6.1.1 Deterministic design base

The safety design of an APWR satisfies, from a deterministic design point of view, the design criteria for design base events. Using probabilistic assessments, the APWR is designed so that it has sufficient margins beyond those required for design base events. The design base events are abnormal operating conditions classified into two groups: abnormal operating transients and accidents during operation. Safety criteria have been set for each group. The standards for radiation exposure are specified for normal operation and for accidents, thus reducing the risk to the general public and to employees to below allowable limits.

6.1.2 Risk reduction

To further reduce risk and provide increased protection, the reactor is designed to have a high degree of safety through simplification, economy, and ease of maintenance. Specifically, it is designed with the following design targets:

- The core damage frequency during power operation should be used as a quantitative index for the increase in safety. As a target, the probability should be reduced to about 1/10 that of the most recent Japanese PWR.

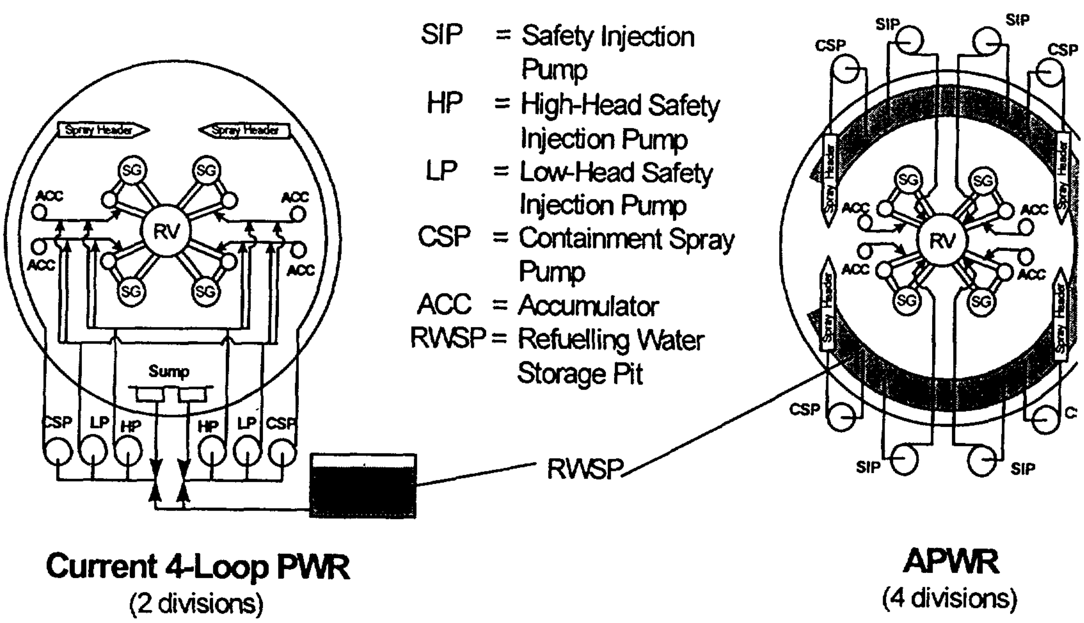

FIG. 4. APWR emergency core cooling systems compared with current PWR.

- The core damage frequency during shutdown should be approximately the same as the target for power operation.
- For further protection, the containment failure probability (CFP) should be reduced to an appropriate level (to approximately 1/10 the core damage frequency, as a target).
- The structure of containment should be designed so that its functions can be maintained as a target for one day during quasi-static pressurization following a severe accident. For premature failure modes caused by missiles and dynamic loads, countermeasures should be taken for equipment, etc.

In particular, these measures are concretely classified as follows.

- Countermeasures against core damage during power operation.
  Although a sufficiently low core damage frequency can be achieved as a result of the increase in safety provided by the four sub-system safety systems, installation of the emergency water source inside the containment, etc., countermeasures against an interface LOCA and other events have been also taken for further risk reduction.
- Countermeasures for increased safety during shutdowns.
  These include installation of an automatic interlock to isolate the letdown line when the RCS water level is lowered, improvement of water level monitoring, improvement of the RCS water injection function during shutdowns and other countermeasures. These countermeasures will reduce possible risks.
- Countermeasures for mitigating the effects of an accident.
  These include the use of the CV air re-circulation systems, alternate sprays supplied from the fire service water systems, countermeasures for hydrogen control, etc., and, at the same time, countermeasures, such as water injection into the cavity from the fire service systems, as well as improvement of the cavity shape, against events that could become a potential threat to the containment.

6.2 Safety systems and their characteristics (active, passive, and inherent)

6.2.1 Safety System Configuration

The primary safeguards system consists of the safety injection system and containment spray system, which are installed in conventional nuclear power plants and for which the equipment has been integrated functionally.

The primary safeguards system consists of four identical and independent mechanical sub-systems. Power is fed from two independent and redundant emergency power systems.

The basic configuration is as follows.

- Four sub-systems, each having one safety injection pump, containment spray/residual heat removal pump, and containment spray/residual heat removal cooler
- One refuelling water pit installed in the containment
- Four advanced accumulator tanks

The advanced accumulators refill the reactor vessel lower plenum and downcomer immediately after a LOCA, with a medium to large break size, then inject water to re-flood the core and function as both the accumulator tank and low-pressure injection pump of a conventional plant. Therefore, the pump that has the function to inject water into the core is the safety injection pump only.

The safety injection pump is connected so that it takes water from the refuelling water pit and feeds cooling water to the reactor vessel. The refuelling water pit is located at the bottom of the containment vessel.

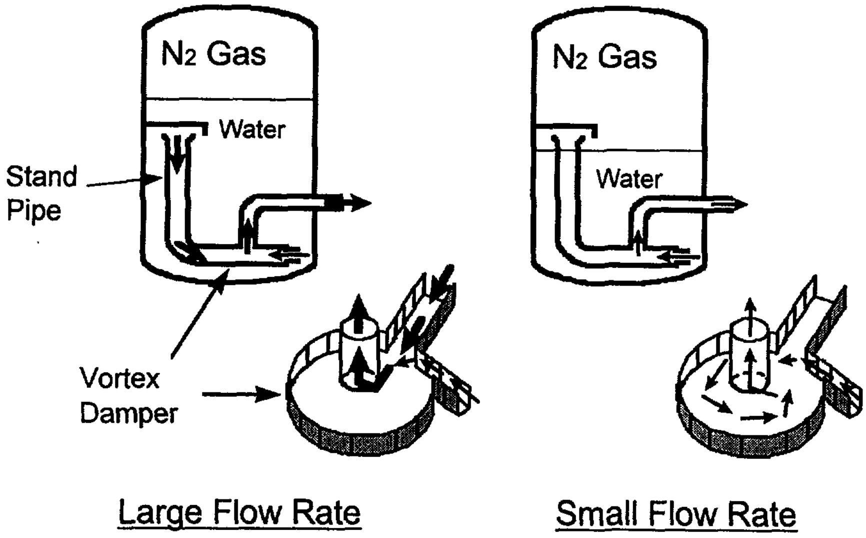

FIG. 5. Outline of the advanced accumulator.

The containment spray/residual heat removal pumps are used as residual heat removal pumps, and also used as containment spray pumps.

The auxiliary feedwater system (AFWS) supplies the auxiliary feedwater to the steam generators when the normal feedwater system is not available. This system consists of two electric auxiliary feedwater pumps and two turbine-driven auxiliary feedwater pumps.

6.2.2 Advanced Accumulator

The advanced accumulator of the APWR injects boric acid water by means of pre-charged nitrogen pressure inside the tank, the same as in the conventional type. The main modification is the installation of a stand pipe and a vortex damper inside the tank to change the injection flow rate by utilizing "fluidics" technology as described below. Figure 5 shows the outline of the advanced accumulator tank and its internal structure.

In the conventional design, accumulators supply cooling water with large flow rate during the early stages and relatively small flow rate injection during the later stages, using both low head and high head injection pumps.

The advanced accumulators inject cooling water into core with large flow rate in a way similar to that of conventional accumulator design during early stages of Large Break LOCA. However, they passively change the injection flow to a relatively smaller rate during the later stages, thus serving the function of conventional low head injection pumps. After core re-flooding is terminated and accumulator tanks become empty, long-term core cooling is performed by high head injection pumps, which have enough capacity to remove decay heat from core.

Figure 6 shows the flow characteristics of the advanced accumulator compared with those of conventional type.

6.2.3 Safety Injection System

The emergency core cooling function of the primary safeguards system is to feed sufficient cooling water into the core in a LOCA.

When a safety injection signal is initiated, the safety injection pumps start automatically and take water from the refuelling water pit located in the containment, injecting cooling water directly into the reactor vessel without passing through the loop.

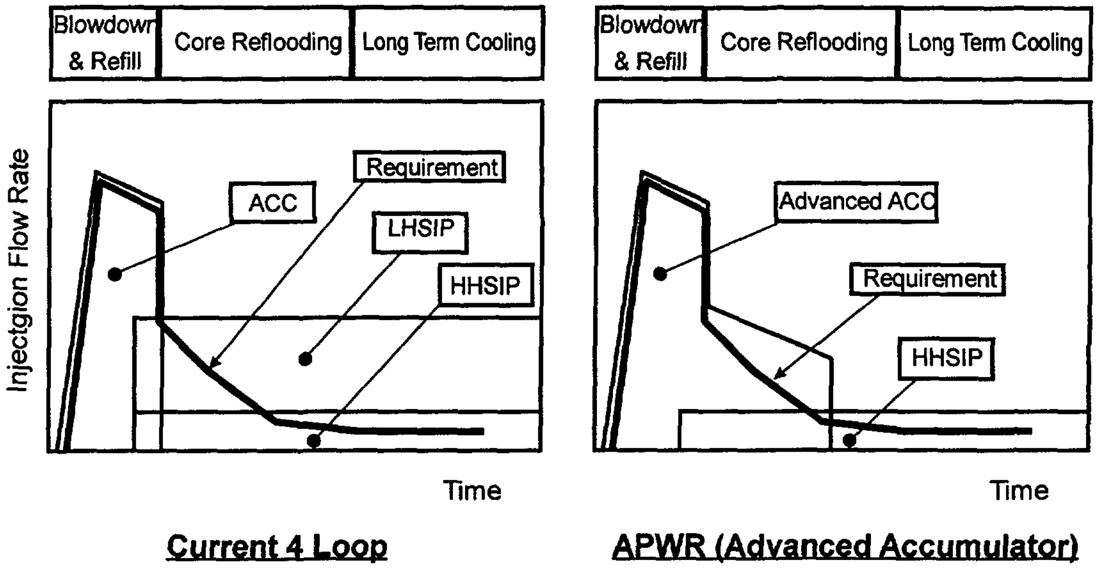

FIG. 6. Flow injection characteristics of the advanced accumulator.

Immediately after the blow-down of the primary coolant during a LOCA caused by a large or medium-sized break, the advanced accumulators are used to refill water into the reactor vessel lower plenum and downcomers, and to inject cooling water until the core is re-flooded. At the start of injection, cooling water is injected at a large flow rate, when the water in an accumulator reaches a certain level, the flow damper changes the flow to a smaller or rate, which provides an injection flow similar to that given by the safety low head injection pumps in current plants.

6.2.4 Containment Spray System

Four containment spray/residual heat removal pumps and coolers function as a containment spray system if a LOCA or main steam line break accident occurs.

When a containment spray is initiated, the stop valves in the pump discharge lines open automatically. The spray/residual heat removal pumps take water from the refuelling water pit and supply it to the containment spray header located at the top of the containment through the spray/residual heat removal coolers.

This system also has a residual heat removal function for removing decay heat from the core in normal cooling of plants and refuelling.

6.2.5 In-Containment Water Storage Pit

The refuelling water storage pit has a horseshoe shape, and is located at the bottom level of the containment.

It provides a continuous source of water for the safety injection pumps and spray/residual heat removal pumps. Therefore, switching from the refuelling water tank to the containment re-circulation sump is not necessary, in contrast to conventional plants.

6.2.6 Auxiliary Feedwater System

The auxiliary feedwater system has two motor-driven pumps and two turbine-driven pumps to increase reliability, when coping with a station blackout.

Upon receiving signals from the reactor protection system, the system starts feeding water automatically from the condensation pit to the steam generators.

6.3 Severe accidents (Beyond design basis accidents)

6.3.1 Prevention of severe accident

In the preliminary design of the APWR a safety system with four sub-systems has been adopted, the RWSP has been located in the containment, advanced accumulators have been included, and the auxiliary feedwater system and auxiliary cooling water system/sea water system have been improved functionally. Thus a high level of safety has already been provided to ensure core integrity.

Regarding the interface LOCA that bypasses the containment, the corresponding parts of the piping in the residual heat removal system have been designed with a higher rating to prevent the interface LOCA from occurring because such an accident can have very serious consequences for the environment.

6.3.2 Countermeasures during plant shutdowns

As safety enhancement countermeasures during mid-loop operating mode, which are especially important during plant shutdowns, the following measures have been adopted: RCS high water level operation, reinforcement of RCS water level indicators, automatic isolation of letdown line due to low RCS water level, reinforcement of water injection functions during lowering of RCS water level, etc., in order to reduce the probability of core damage.

In addition, as a precaution against the event of abnormal dilution of boric acid during an external power failure, interlock to prevent boron dilution has been provided.

6.3.3 Mitigation of severe accidents

In an APWR plant, as shown in section 6.1, mitigation of the consequences of a severe accident is also considered from the viewpoint of risk reduction and further protection. Specifically, as shown below, countermeasures against those events that threaten the integrity of the containment vessel (CV) are under consideration.

As countermeasures against debris dispersion, enhancement of the depressurization function of the primary system and improvement of the RV cavity form has been realized. As countermeasure against damage by quasi-static over-pressure, the normal CV air re-circulation system and an alternative CV spray supplied from the fire service water system can be used. These systems can be used to cool the CV and reduce pressure if the CV spray system is not available.

As a countermeasure against CV damage due to hydrogen combustion, a hydrogen control system (igniters) will be installed to control the hydrogen concentration.

To provide adequate cooling of molten debris ejected from the reactor vessel, sufficient floor space will be provided in the reactor vessel cavity and water will be injected into the cavity from the fire service water system. A 1 m thick protective wall of concrete will also be provided, so that the CV boundary is not exposed directly to the debris. Thus, the molten debris will be coolable, and erosion of the concrete and overheating of the CV atmosphere prevented.

As countermeasures against the dispersion of debris, further enhancement of depressurization function and improvement of the reactor vessel cavity form are being pursued. It is also considered that the outlet from the reactor vessel cavity to the other CV spaces should be constructed like a labyrinth, in order to prevent direct containment heating (DCH) or direct contact of the debris with the CV boundary.

7. PLANT LAYOUT

The Plant must be laid out so that the safety of reactor facilities is not impaired, and the exposure dose around the plant is below a specified limit. Also separation of redundant trains, earthquake resistance, and maintenance of the safety system equipment must be considered to give an optimum arrangement.

7.1 Buildings and structures, including plot plan

Figure 7 shows the arrangement of an APWR plant. The standard arrangement is for a twin unit plant consisting of two reactor buildings and a common control building, waste building, and turbine building.

7.1.1 Design requirements

The buildings, structures, equipment, and pipes are classified into the following three classes: A, B, and C. The seismic design must be made according to the class concerned.

Class A: Those facilities which contain radioactive materials themselves or are associated directly with facilities which contain radioactive materials, and which may release radioactive materials outside the plant if they fail to function properly. Also, facilities which are required to prevent such releases and to reduce the effect of radioactive materials dispersed to the environment if an accident occurs, and those which have serious consequences for the plant.

Class B: Those that have relatively smaller effects on the plant compared to Class A.

Class C: Those for which safety standards equivalent to those of general industrial facilities are adequate and facilities not classified as A or B.

Based on the above classifications, the seismic design of the buildings is made by classifying the reactor building and control building into Class A, the waste building into Class B, and the turbine building into Class C.

To ensure safety against aircraft impacts, in principle, a site must be selected which is not close to an airport and air route if aircraft impacts are not to be considered in the design. However, if the location of the plant makes consideration of aircraft impacts unavoidable, the aircraft impact conditions must be determined properly, and necessary countermeasures must be taken.

The plant must be designed as follows for internal and external events such as jet aircraft, missiles, and fires.

Jet aircraft and missiles: The design must be made in such a way that the safety of the reactor is not impaired due to the effects of missiles and broken pipes.

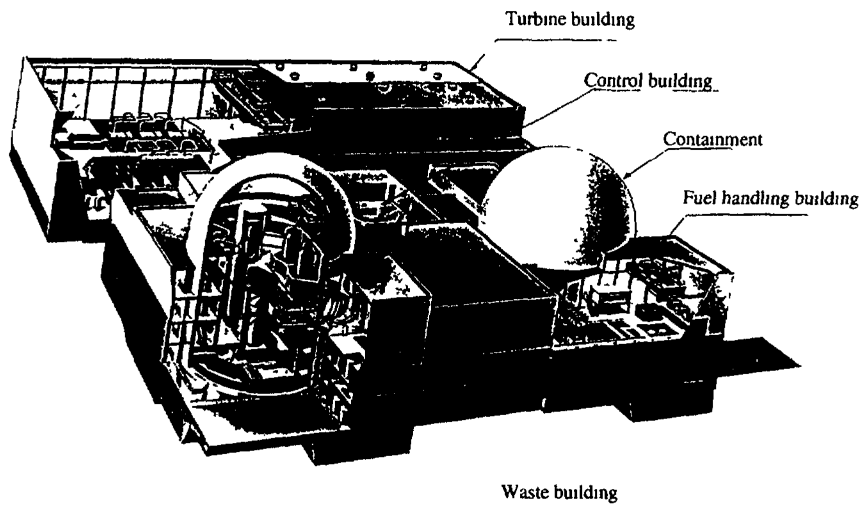

FIG. 7. APWR – General view of buildings.

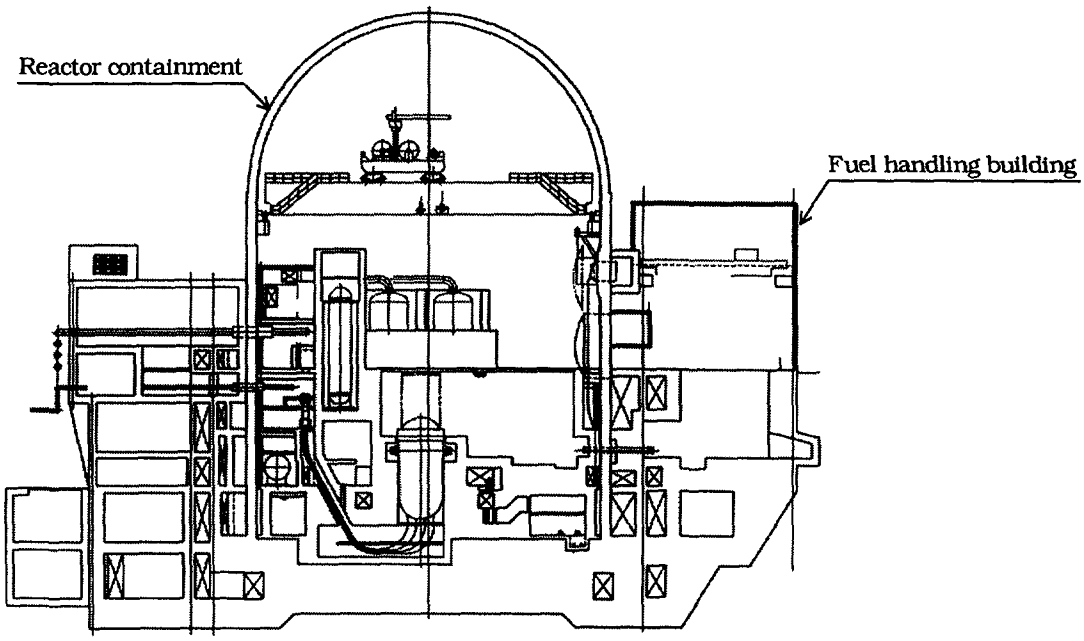

FIG. 8. APWR — Reactor building cross-section.

Fire: To prevent the safety of reactor facilities from being impaired by fire, the plant must be designed using a proper combination of three general rules based on the Japanese "Guidance for Verification of Fire Protection of LWR Facilities for Power Generation."

a. Prevention of fires

b. Detection of fires and fire extinguishing

c. Reduction of the effects of fires

In principle, the structures, systems, and equipment critical for safety must be so designed that the reactor facilities do not make common use of any one of them provided that, judging from the functions and construction, it is determined that the safety of the reactor may be impaired by such common use.

The reactor facilities must be arranged in the plant site of the plant so that the exposure dose to the general public in those areas outside the controlled areas and around the plant is below a specified limit. Furthermore, they must be sufficiently far from the site boundary so that the exposure dose received in areas outside the site during severe accidents and hypothetical accidents is adequately below the target dose indicated in the Japanese "Guidance for Verification of Reactor Siting."

The interior of the plant must be divided into zones according to the radiation levels, and suitable radiation shielding must be provided.

7.2 Reactor building

The reactor building consists of the reactor containment facility and the peripheral buildings in which the fuel handling systems and associated systems are installed. Figure 8 shows the cross-section of the reactor building.

7.3 Containment

The containment is part of the reactor containment facility and includes the internal concrete and the annulus. The reactor containment facility is part of the engineered safety systems, which includes the emergency core cooling system, the containment spray system, and annulus air purification system.

The containment system is designed to suppress or prevent the possible dispersion of the large quantities of radioactive materials which would be released if extensive fuel failures were to occur in the reactor resulting from damage or failure of the reactor facilities such as the primary cooling system, main steam system, and feedwater system.

The leakage preventing function of the containment is provided by a 6.4 mm thick steel liner on the inner surface while the pressure withstanding function is provided by the concrete structure. An enclosed space (annulus) is formed around the lower part of the containment shell to provide a double containment and the containment penetrations for pipes, cables, ducts, and air locks pass through the annulus.

The containment is designed so that the leak-rate is less than 0.1% per day of the weight of air in the containment at a pressure of 0.9 x maximum design air pressure at normal temperatures.

At this containment leak-rate, the general public will not be affected by radiation even if the facilities related to the primary coolant system fail or are damaged. Therefore, severe accidents must also be carefully considered to ensure the integrity of the containment.

In current-day plants, the refuelling water, which is the water supply used after an accident, was stored in a tank outside the containment. In an APWR, however, in order to avoid a failure to switch over the water source from the tank to the recirculation sump inside the containment, the refuelling water is stored in a pit inside the containment.

Also, a proper space has been provided below the reactor vessel so that debris will not concentrate in one location if a hypothetical ejection of molten debris occurs, and the space is shaped to catch the debris easily to prevent it from being splashed, as far as possible, into the general spaces of the containment.

Although the containment is designed to withstand the maximum design pressure and temperature determined from the design basis events, there will be no excessive leakage under the pressure and temperature conditions expected during severe accidents. This must be checked to make sure.

7.4 Turbine building

The turbine generator, condensate and feedwater system auxiliary equipment, and other equipment are installed in the turbine building. The turbine generator is arranged with its axis in line with the reactor.

The foundation of the turbine building is made of concrete to reduce the thickness of the mat. The floor of the turbine building below ground level is made of concrete, and the floors above ground are steel structures that are designed to withstand all loads including the load of the overhead travelling crane.

The turbine generator systems are arranged so that the space can be utilized effectively not only during the construction of the plant but also during operation and periodical inspections.

Suitable spaces have been provided for inspection access, transportation of tools for inspections and maintenance, and disassembly in a way that reduces the volume of the building.

7.5 Other buildings

The buildings and systems have been arranged so as to optimize the relation between the systems, separation of safety system equipment, seismic resistance, maintenance, etc.

EP1000 PASSIVE PLANT DESCRIPTION

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Abstract

In 1994, a group of European Utilities, together with Westinghouse and its Industrial Partner GENESI (an Italian consortium including ANSALDO and FIAT), initiated a program designated EPP (European Passive Plant) to evaluate Westinghouse Passive Nuclear Plant Technology for application in Europe [1, 2]. In Phase 1 of the European Passive Plant Program which was completed in 1996, a 1000 MWe passive plant reference design (EP1000) was established which conforms to the European Utility Requirements (EUR) [3] and is expected to meet the European Safety Authorities requirements. Phase 2 of the program was initiated in 1997 with the objective of developing the Nuclear Island design details and performing supporting analyses to start development of Safety Case Report (SCR) for submittal to European Licensing Authorities. The first part of Phase 2, "Design Definition" phase (Phase 2A) will be completed at the end of 1998, the main efforts being design definition of key systems and structures, development of the Nuclear Island layout, and performing preliminary safety analyses to support design efforts [4, 5, 6]. The second part, "Phase 2B", includes both the analyses and evaluations required to demonstrate the adequacy of the design, and to support the preparation of Safety Case Report. The second part of Phase 2 of the program will start at the beginning of 1999 and will be completed in the 2001. Incorporation of the EUR has been a key design requirement for the EP1000 from the beginning of the program. Detailed design solutions to meet the EUR have been defined [4, 6, 7] and the safety approach has also been developed based on the EUR guidelines. This paper integrates and updates the plant description reported in the IAEA TECDOC-968 [8]. The most significant developments of the EP1000 plant design during Phase 2A of the EPP program are described and reference is made to the key design requirements set by the EUR Rev. B document.

1 INTRODUCTION

In 1994, a group of European Utilities, together with Westinghouse and its Industrial Partner GENESI (an Italian consortium including ANSALDO and FIAT), initiated a program designated EPP (European Passive Plant) to evaluate Westinghouse Passive Nuclear Plant Technology for application in Europe [1, 2]. The initial European utility group included the following organizations:

- Agrupación eléctrica para el Desarrollo Tecnologico Nuclear (DTN), Spain
- Electricité de France, France
- ENEL, SpA., Italy
- Imatran Voima Oy, Finland
- Scottish Nuclear Limited (acting for itself and on behalf of Nuclear Electric plc), U.K.
- Tractebel Energy Engineering, Belgium
- UAK (Represented by NOK-Beznau), Switzerland
- Vattenfall AB, Ringhals, Sweden

In Phase 1 of the European Passive Plant Program, which was completed in 1996, a 1000 MWe passive plant reference design (EP 1000) was established which conforms to the European Utility Requirements[1] (EUR) [3] and is expected to meet the European Safety Authorities requirements.

1 The EUR (European Utility Requirements) effort was launched in December 1991 by five European Utilities, later joined by four others. The main objective of the EUR organization is to produce a common set of utility requirements, endorsed by major European utilities for the next generation of Light Water Reactor (LWR) nuclear power plants.

The base design for this passive plant program is the AP-600. The passive PWR approach to design is to strike a balance between the use of proven technology and new concepts - the advantage of the traditional Westinghouse PWR technology combined with natural, passive safety systems. The result is a greatly streamlined plant that can meet safety regulations and reliability requirements, be economically competitive and promote broader public confidence in nuclear energy.

With respect to safety systems and containment, the reference plant design closely follows that of the Westinghouse SPWR design [9, 10], while the AP-600 plant design has been taken as the basis for the EP 1000 reference design in the auxiliary system design areas. However, the EP 1000 design also includes features required to meet the EUR, as well as key European licensing requirements [4, 6].

The ultimate objective of Phase 2 of the program is to develop design details and perform supporting analyses to produce a Safety Case Report for submittal to European Safety Authorities. The first part of Phase 2, hereafter referred as Phase 2A, started at the beginning of 1997 and will be completed at the end of 1998. Scope of this phase of the program is to focus on improving the design of important systems and structures.

In parallel to the Phase 2A effort, a group of European Utilities are sponsoring the activities for the preparation of the EP1000 EUR Volume 3. Volume 3 will be the EP1000 plant example and compliance assessment against the EUR. The EP1000 EUR Volume 3 program began in June 1997 and will be concluded at the end of 1998.

The EP 1000 has a well-defined design basis that is confirmed through engineering analyses and testing and is in conformance with the EUR. Some of the high level design characteristics of the plant are [1, 8]:

- The standard design will be applicable to European sites.
- The net electrical power is 1000 MWe and the thermal power 2910 MWt.
- The average coolant temperature at the reactor vessel exit does not exceed 617 °F (325 °C) during normal operation even with 10% of the steam generator tubes plugged.
- The reactor core is a low power density core that uses the Westinghouse 12 ft. (3658 mm), 17x17 fuel assembly.
- Short lead time and construction schedule are expected.
- No plant prototype is needed since proven power generating system components are used
- Safety systems are passive; they require no operator action for more than 24 hours after an accident, and maintain core and containment cooling for a protracted period of time without AC power.
- Predicted core damage frequency is below 10E-5 /yr., and frequency of significant release is below 10E-6/yr.
- Occupational radiation exposure is expected to be below 0.5 man-Sv/yr. (50 man-rem/yr.)
- The core is designed for a 18-24 month fuel cycle assuming 87 % capacity factor.
- Two preliminary 24 month UO_2 (20,9 EFPM) and an 18 month 50 % MOX Core (15,7 EFPM) have been designed to cope with the Low Boron Capability and MOX core requirements of the EUR.
- Refuelling outages without major problems or major maintenance will be conducted in 17 days or less.
- Plant design aims at a lifetime of 60 years without replacement of the reactor vessel.
- The design aims at an overall plant availability greater than 90 %, including forced and planned outages; the goal for unplanned reactor trips is less than one per year.
- The design incorporates the US advanced light water reactor (ALWR) utility requirements and satisfies the US general design criteria of 10 CFR 50, Appendix A.

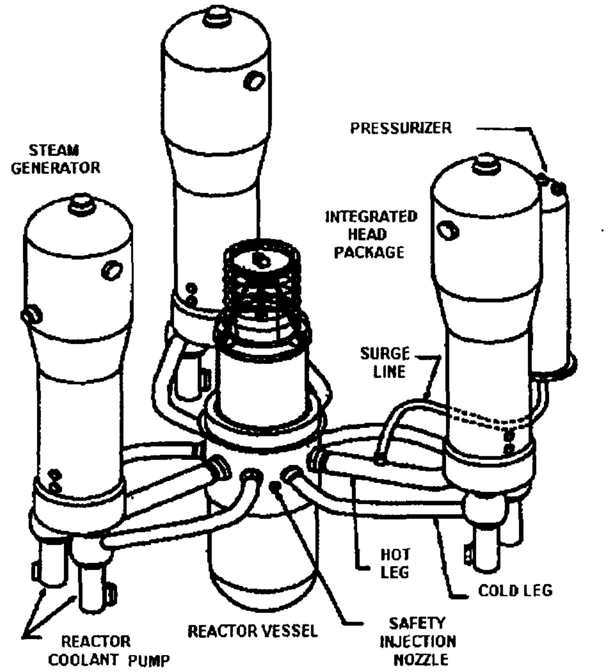

FIG. 1. EP 1000 Reactor coolant system layout

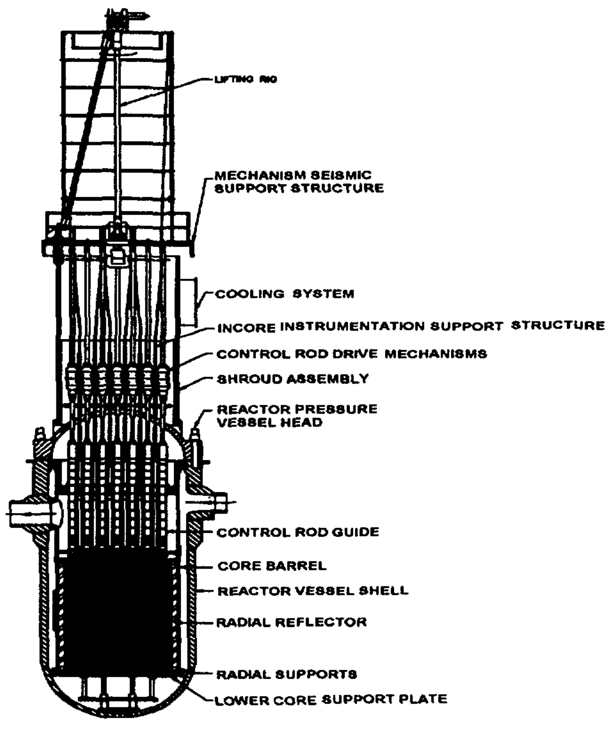

FIG. 2. EP 1000 reactor system

2 DESCRIPTION OF THE NUCLEAR SYSTEMS

2.1 Primary circuit and its main characteristic

An isometric view of the main loop is shown in Figure 1. The coolant loops consist of three hot leg and six cold leg pipes and the reactor coolant pumps (RCPs) are mounted directly on the channel head of each steam generator. The integration of the pump suction into the bottom of the steam generator channel head eliminates the cross-over leg of coolant loop piping, thus avoiding the potential for core uncovery due to loop seal venting after a small loss-of-coolant accident (LOCA).

During Phase 2A of the EPP program several design activities have been performed to define the details of the Reactor Coolant System (RCS). Most of the modifications to the initial design were made following AP-600 and SPWR latest development while others have been developed specifically for the EP 1000 plant with special attention to possible cost reduction areas.

In particular, based on the results of the optimization study of the Automatic Depressurization System, the following main modifications have been defined:

- Elimination of the Third Stage ADS Valves;
- Modification of the Final ADS Stage to include three 10-in squib valve flow paths for each of the two final ADS stage valve group;
- Reduction of the surge line from 18 to 16 inches;
- Reduction of the Pressurizer Safety Valve inlet piping from 14 to 10 inches.

Additional modifications that were implemented are:

- Addition of a Pressurizer Nitrogen Fill Connection to perform a two-phase cool-down;
- Modification of the Vessel Head Vent design according to EUR Safety Class F2 system.

2.2 Reactor core and fuel design

The core, reactor vessel, and reactor internals of the EP 1000, shown in Figure 2, are similar to those of currently operating Westinghouse PWR plants, but several new features are incorporated to enhance the performance characteristics, as compared with existing plants, and meet the EUR.

2.2.1 Core Design

The reactor core uses the Westinghouse 12 foot (3658 mm), 17x17 fuel assembly. A low power density is achieved by making the core larger than previous 1000 MWe designs, with the number of fuel assemblies increased from 157 to 193. This configuration results in core power density and average linear power density enhancements of about 25 percent, to 88,84 kW/l and 15,16 kW/m, over existing plants of the same power rating. This results in lower fuel enrichments, less reliance on burnable absorbers, and longer achievable operating cycles.

Another core design feature is the use of reduced-worth control rods (termed "gray" rods) to achieve daily load-follow capability without requiring daily changes in the soluble boron concentration [1].

A total of 81 RCCAs are incorporated in the core design (61 black RCCAs and 20 Gray RCCAs). The RCCA absorber material is Ag-In-Cd.

The core is surrounded by a stainless steel radial neutron reflector, which contributes to lowering fuel cycle cost and to reduce neutron fluence on the reactor vessel wall, an important factor in view of the 60 year lifetime design objective.

For the initial core design, discrete Wet Annular Burnable Absorbers (WABA)[2] rodlets, which are consolidated into burnable absorber assemblies, and Integral Fuel Burnable Absorbers (IFBAs) are used. Discrete absorber designs, integral fuel burnable absorber designs, or both, may be used in subsequent reloads (IFBAs are the preferred burnable absorber design).

Core physics characteristics have been evaluated for a 24 month operating cycle assuming a capacity factor of 87 percent. This results in a cycle energy requirement of approximately 635 Effective Full Power Days (EFPD).

Two preliminary 24 month cycle UO_2 fuel management schemes have been developed for the EP 1000 reactor as part of the Phase 2A program. These schemes are required to meet the EUR Rev. B Low Boron Design requirements, as well as all applicable conventional safety analysis limits. One design uses mid-enriched (2,60 weight percent [w/o]) axial blankets that provide an economic benefit equivalent to approximately a 0,1 w/o reduction in feed fuel average enrichment, when compared to the second design which uses a single uniform axial fuel enrichment.

The only observed consequence resulting from the use of these mid-enriched axial blankets is an acceptable increase in base load steady state axial peaking factor (F_Z), and the nuclear heat flux hot channel factor, (F_Q). The nuclear enthalpy rise hot channel factor, $F_{\Delta H}$, during steady state operation is, however, comparable for both designs. Both designs assume that a stainless steel radial reflector design (similar to the one utilized for the AP-600 and SPWR) is employed.

In addition to the reference core configuration, a preliminary design for a 50 % MOX, Low Soluble Boron design core has been developed during the Phase 2A of the EPP Program.

While the MOX core design exhibits an acceptable core power distribution behaviour both radially and axially, additional activities are still anticipated to optimize the location of the Axial Offset Banks in order to limit their insertion in the core, improve their efficiency and make them more effective with the MSHIM control strategy.

For both the UO_2 and MOX design, the following design requirements are meet:

- The moderator temperature coefficient (MTC) at hot full power (HFP), no xenon, is more negative than -13 pcm/°F, through 100 percent of the fuel cycle. This ensures compliance with the EUR low boron requirement to avoid core and RCS pressure boundary damage after an ATWS event through 100% cycle life.
- The core is supercritical by no more than 7.0 percent ($k_{eff} < 1.070$) through 100 percent of the fuel cycle (including no xenon), at zero boron conditions. This ensures compliance with the EUR low boron requirement to avoid core and primary system damage following rapid boron dilution events.
- The core is sub-critical ($k_{eff} < 1.00$) at no-load temperature and rated pressure, assuming no fission power, all control/shutdown rods inserted, no xenon, and no soluble boron in the coolant. This ensures compliance with the EUR low boron requirement to assure sub-criticality at Hot Zero Power (HZP).

2.2.2 Reactor Vessel Internals

The reactor vessel internals design has been reviewed during Phase 2A as a consequence of the activities related to Design Extension Conditions as required by EUR. The ability of the EP1000 to provide in-vessel retention and cooling of core debris following design extension conditions was evaluated as part of the In-Vessel Debris Retention Study. The study identified a problem with in-

[2] The WABA rodlets are located in selected guide thimbles of the fuel assemblies. The WABA absorber material consists of a thin walled alumina (Al_2O_3) pellet containing B-10 in a boron carbide (B_4C).

vessel coolability and the need for a modification to the reactor vessel lower internals configuration was subsequently identified to resolve the issue. The bottom of the EP1000 lower core support plate sits higher in the reactor vessel than in the reference plant design (AP600) and the additional metal mass of the support plate and reflector are not submerged into the molten metal pool. The resulting melt geometry produces a heat flux profile that exceeds the critical heat flux for the reactor vessel and the reactor vessel fails. A revised configuration, that places the bottom of the core support plate approximately 10 inches (260 mm) lower, allows contact between the lower support plate and the molten debris pool. The increased thickness of the molten metal layer spreads the heat transfer over a larger area of the reactor vessel and reduces the heat flux below the critical heat flux. This preliminary design change to the lower internals proposed to address the concern with in-vessel coolability and retention of core debris will be further evaluated as part of the Phase 2B design activities.

2.3 Reactor auxiliary systems

Some major auxiliary systems of the EP 1000 discussed in the following are:

- Auxiliary Cooling Systems
- Spent Fuel Pool Cooling System
- Boron Recycling System
- Secondary Containment Ventilation System (PAFS)

The major modifications to the above systems, with respect to the initial configuration derived from the SPWR and AP-600 designs as reported in the IAEA-TECDOC-968 [8], will be described and the reason for modification outlined. For description of remaining auxiliary systems, reference is made to IAEA-TECDOC-968 [8].

2.3.1 Auxiliary Cooling System Design

The EP 1000 Auxiliary Cooling Systems include:

- Normal Residual Heat Removal System (RNS)
- Component Cooling Water System (CCS)
- Service Water System (SWS)
- Spent Fuel Pool Cooling System (SFS)

EUR criteria that have impacted the design of the EP 1000 heat removal systems include:

- Use of MOX fuel
- Boron Recycling (waste evaporator heat loads)
- More rapid plant cool-down times
- Site parameters (e.g., max/min ultimate heat sink temperatures)
- Increase spent fuel storage capability
- Aggressive EP1000 refuelling schedule that requires full core off load at about 108 hours to be able to meet the EUR 17 day refuelling requirement.

The limiting EUR performance requirements for sizing of the RNS and CCS are:

- The Plant should be capable of shutdown from Hot Zero Power to cold shutdown at a temperature less than 60 °C (140 °F) within 20 hours (EUR 2.2 -2.1).
- The RNS shall have sufficient capacity to bring the reactor to 90 °C (194 °F) within 36 hours after shutdown with a single failure in the RNS (EUR 2.8 - 2.4.1.3.1).
- Initiation of system (RNS) operation 6 hours after reactor shutdown (EUR 2.8 - 2.4.1.3.1).

The above requirements have resulted in increased RNS/CCS capability. Larger CCS heat exchangers have been incorporated to limit the size of the RNS heat exchangers. In addition, RNS heat exchanger heat transfer effectiveness has been increased via use of a two-shell pass heat exchanger design versus a one-shell pass design.

2.3.2 Boron Recycle

The EUR requires that the boron utilized in the reactor plant be recycled (EUR 2.8.1.2.4.1). The EP 1000 Chemical and Volume Control System (CVS) and Liquid Radwaste System (WLS) designs have been modified to accommodate boron recycling.

The WLS [4, 6] design includes an evaporator-degasifier with the duel function of providing 1) degassing of the primary coolant, and 2) boron recovery. The evaporator-degasifier is designed to operate in a semi-continuous mode where vapour is continuously drawn while the solution to be concentrated is continuously fed to the evaporator. Effluent for reprocessing is received from the CVS and the reactor coolant drain tank. The concentrated boric acid is drawn from the evaporator in a batch mode when the desired concentration is reached. The EP 1000 evaporator-degasifier concentrates boric acid to 4.0 weight percent solution at a rate of 12 m^3/hr. In the degassing mode, the evaporator-degasifier processes 23 m^3/hr of primary coolant received from the CVS.

Modifications have also been made to the CVS. An additional boric acid tank (BAT) has been implemented in the design. Fresh recycled concentrate from the WLS evaporator is routed to one of the two BATs where the boron concentration is continuously monitored. The other BAT is used as the boron source for the primary system. A side benefit of the recycling process is the generation of high quality condensate that can be reused as reactor makeup water. The processed water contains tritium, and therefore, is not acceptable for diversion into the Demineralized Water Tanks. As a result, Reactor Makeup Water Tanks have been added to the EP1000 CVS design to store the recycled reactor makeup water prior to reuse by the CVS for reactor makeup.

2.3.3 Spent Fuel Pool Cooling System (SFS) Design

The EP 1000 SFS and spent fuel pool (SFP) have been sized based on the following [4]:

- Accommodation of 15 years of MOX spent fuel, plus 10 years of UO_2 spent fuel, plus 1 full core offload (EUR 1.5.7.1)
- full core off load starting at 100 hours after shutdown and completed at 148 hours (EUR 2.8.0.5.2)
- SFP temperature < 50°C (122°F) at all times except in incident or accident conditions when one train is not available temperature shall be < 70°C (158°F). (EUR 2.8.2.10.4.3)
- MOX fuel (MOX decay heat at 15 years can be three times higher than UO_2) (EUR 2.2.3.1)
- EP 1000 goal of completing full core offload at 108 hours after shutdown to support the EUR 17 day refuelling outage requirement. (EUR 2.2.7.2.2)

To ensure SFP cooling for at least 72 hours following a seismic event, a dedicated line from the Passive Containment Cooling System (PCS) water storage tank to the SFP has been installed, Figure 3.

The PCS water source is manually actuated following a seismic event that results in loss of normal SFP cooling.

2.3.4 Secondary Containment Ventilation System (PAFS)

The Secondary Containment Ventilation System (Passive Annulus Filtration Systems - PAFS) is part of the EP 1000 Secondary Containment Ventilation Systems (VGS). It is designed to function following a severe accident to limit the offsite dose. The EP 1000 is equipped with a secondary containment (cf. Reference [4] for a description of the secondary containment and associated functions and requirements). For the steel containment reference configuration, the secondary containment is

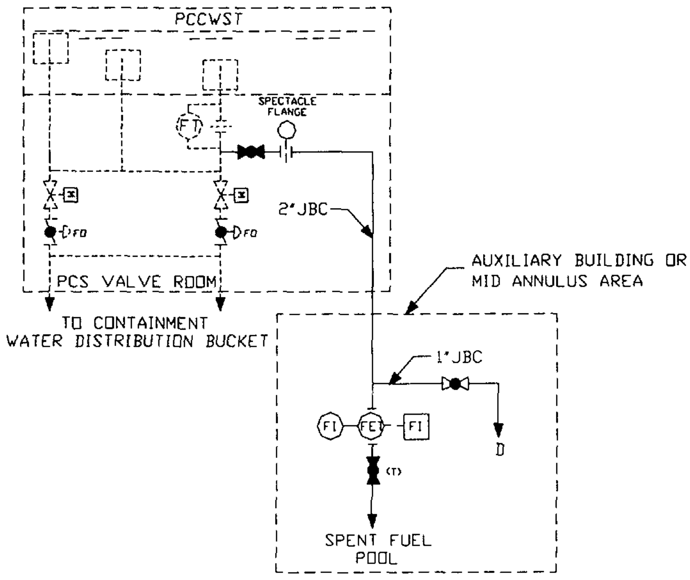

FIG. 3. EP 1000 Spent Fuel Pool Makeup from the PCCWST

defined as the structure that confines the penetration area to collect leakages through the penetrations that constitute the major source of containment leakage.

The PAFS is designed to perform the following major functions:

- Contribute to the limitation of the offsite dose to the value defined by site parameters; and
- Maintain a negative pressure in the annulus penetration (secondary containment).

Initial evaluations indicate the EP1000 radioactive releases to the environment will be low, in compliance with the EUR safety targets for Design Basis Accidents (DBA) without operation of the PAFS. Therefore, the PAFS is provided to fulfill a level F2 safety function which in the EUR Rev. B

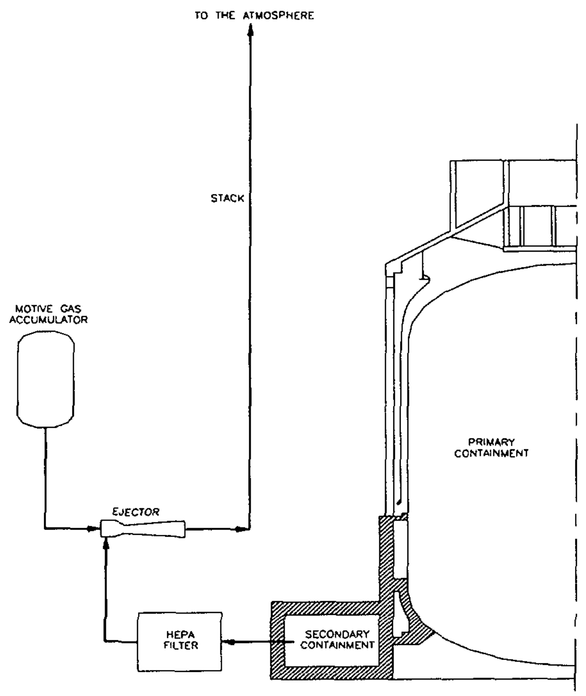

FIG. 4. EP 1000 Passive Annulus Filtration System

is defined as, "to ensure that the releases are kept within the targets set for DEC (design extension conditions)".

The PAFS, figure 4, is connected to the middle and lower annulus of the secondary containment. It consists of two mechanical trains of equipment. Each train consists of one HEPA filter, one ejector and a compressed air storage. The motive force of the ejector is the compressed air stored in tanks, having a capacity per train to support the function of the PAFS for the first 24 hours after a DEC accident. The capacity of both trains, used one after the other, should be able to perform the PAFS function for a period of 72 h.

3 SAFETY CONCEPTS

3.1 Safety requirements and design philosophy

3.1.1 Safety systems and features (active, passive and inherent)

The EP 1000 uses passive safety systems to further enhance plant safety and to satisfy U.S. NRC safety criteria and EUR requirements. The use of passive safety systems has provided significant and measurable improvements in plant simplification, safety, reliability, and investment protection. The passive safety systems require no operator actions to mitigate design basis accidents. These systems use only natural forces such as gravity, natural circulation, and compressed gas to make the systems work; no pumps, fans, diesels, chillers, or other active equipment are used. A few simple valves align and automatically actuate the passive safety systems. To provide high reliability, these valves are designed to actuate to their safe positions upon loss of power or upon receipt of a safeguards actuation signal. However, they are also supported by multiple, reliable power sources to avoid unnecessary actuations.

The EP 1000 passive safety-related systems include:

- The passive core cooling system (PXS);
- The passive containment cooling system (PCS);
- The main control room habitability system (VES);
- Containment isolation.

A brief overview of the first two systems is provided in the following.

3.1.2 Passive Core Cooling System.

The PXS protects the plant against reactor coolant system (RCS) leaks and ruptures of various sizes and locations. The PXS provides the safety functions of core residual heat removal, safety injection, and depressurization. Safety analyses (using U.S. NRC-approved codes) demonstrate the effectiveness of the PXS in protecting the core following all sizes of RCS break events. Even for breaks as severe as the 8-inch vessel injection lines the core remains covered. The PXS provides approximately a 400°F margin to the maximum peak clad temperature limit for the double-ended rupture of a main reactor coolant pipe.

Safety Injection and Depressurization. The PXS, see Figure 5, uses four passive sources of water to maintain core cooling through safety injection. These injection sources include the core makeup tanks (CMTs), the core reflood tanks (CRTs), the accumulators, and the in-containment refuelling water storage tanks (IRWST). These injection sources are directly connected to two nozzles on the reactor vessel so that no injection flow can be spilled for the larger break cases.

The CMTs provide makeup at any RCS pressure, using gravity to provide injection flow. These tanks are located inside the containment, above the RCS loops. The CMT pressure is equalized with the RCS through a line connecting the top of the CMTs to the RCS. The elevation head is sufficient to

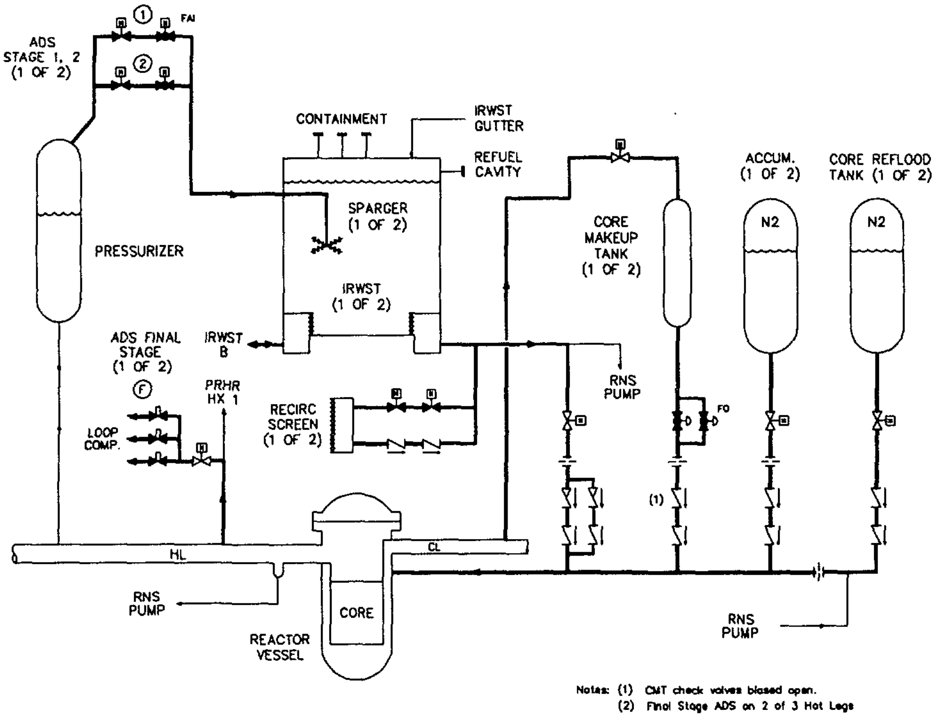

FIG. 5. Passive Core Cooling System Simplified Sketch

overcome the small pressure loss in the injection line. The CMTs are normally full of borated water and isolated by two parallel "fail-open" air-operated globe valves. The isolation valves open when the RCS pressure or level drops to abnormal levels. The tanks are sized to provide extended makeup to the RCS in the case of minor leakage.

For larger leaks, additional water is provided by the accumulators and core reflood tanks (CRTs) which inject water pressurized by compressed nitrogen. The two accumulator tanks operate passively when the RCS pressure drops below the normal gas pressure of 700 psig. The two CRTs operate passively when the RCS pressure drops below the normal gas pressure of 200 psig. The gas pressure forces open check valves that normally isolate the accumulators and CRTs from the RCS. The accumulators and CRTs are sized to respond to the complete severance of the largest RCS pipe, by rapidly refilling the vessel downcomer and lower plenum.

Long-term injection water is provided by gravity from the IRWST, which is located in the containment just above the RCS loops. Normally, the IRWST is isolated from the RCS by check valves. This tank is designed for atmospheric pressure, and the RCS must be depressurized before injection can occur. The depressurization of the RCS is automatically controlled to reduce its pressure to about 12 psig; at which point, the head of water in the IRWST overcomes the low RCS pressure and the pressure loss in the injection lines. The PXS provides for depressurization using its automatic depressurization system (ADS), composed of three stages to permit a relatively slow, controlled RCS pressure reduction. The first three stages are connected to the pressurizer and discharge through spargers into the IRWST. The three fourth-stage depressurization paths are connected to the hot legs, and discharge through redundant isolation valves to the containment. The initial ADS stages are actuated by the CMT water level, using 2-out-of-4 logic to ensure reliability and to prevent spurious actuations. All of the valves use existing nuclear-grade valve body and operator designs, extensively tested to ensure operability in this application.

During a LOCA, the initial volume of water in the IRWST provides injection for at least 6 hours. As it empties, the containment water level rises above the RCS loop level. This level is sufficient to force water to drain through a screen and check valves back into the RCS where it is

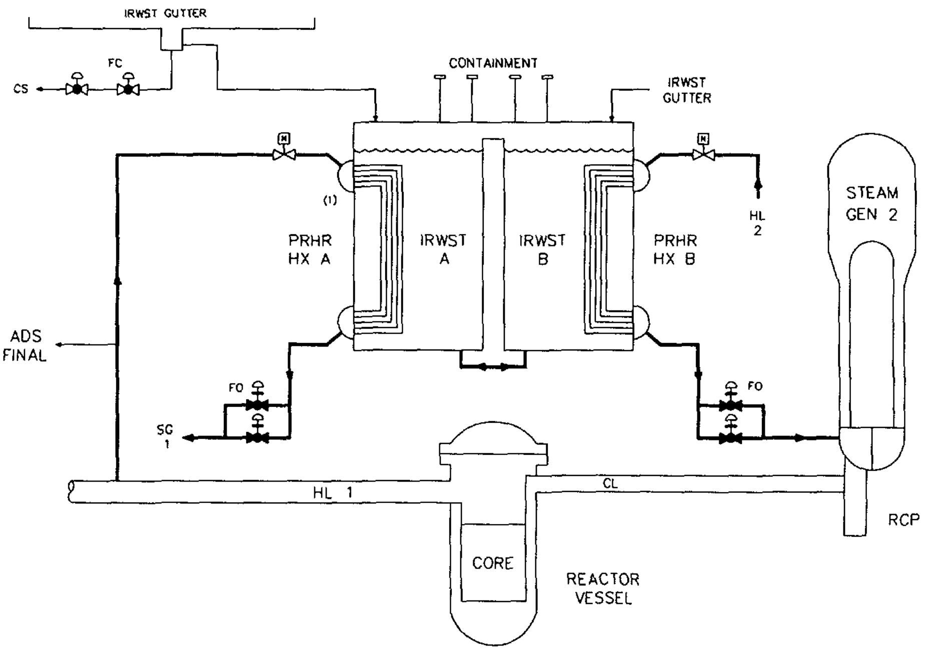

FIG. 6. Passive Residual Heat Removal System Simplified Sketch

turned into steam by core decay heat. The steam is vented to the containment through the ADS valves and the break where it condenses on the inside of the steel containment vessel. The condensate drains down into the IRWST and again becomes available for injection into the RCS.

Passive Residual Heat Removal. The PXS includes two identical passive residual heat removal heat exchangers (PRHR HXs), which are connected through inlet and outlet lines to two of the RCS loops. The PRHR HXs protect the plant against transients that upset the normal steam generator feedwater and steam systems, and satisfy the U.S. NRC safety criteria for loss of feedwater, and feedwater and steam line breaks using single failure assumptions approved by U.S. NRC safety analysis codes.

The IRWST provides the heat sink for the PRHR HXs. The IRWST water volume is sufficient to absorb decay heat for more than 1 hour before the water begins to boil. Once boiling starts, steam passes to the containment where it condenses on the inside of the steel containment vessel and, after collection, drains by gravity back into the IRWST. The PRHR HXs, along with the passive containment cooling, provides indefinite decay heat removal capability with no operator action required. The operator is provided with the capability of controlling the PRHR HX rate such that the RCS temperature can be controlled, if desired. This capability also allows for control of the rate of RCS cool-down.

3.1.3 Passive Containment Cooling System.

The passive containment cooling system (PCS) provides the safety-related ultimate heat sink for the plant. As demonstrated by computer analyses and extensive test programs, the PCS effectively cools the containment following an accident such that the design pressure is not exceeded and the pressure is rapidly reduced. For single steel containment, the steel containment vessel itself provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Steel containment vessels of similar size have been used on operating PWRs. Heat is removed from the containment vessel by a continuous natural circulation flow of air. During an accident, the air cooling is supplemented by evaporation of water. The water drains by gravity from a tank located on top of the containment shield building. Two normally closed fail-open butterfly valves are opened to initiate the water flow. The water tank is sized for 72 hours of operation, after which time the tank is expected to be refilled so that the low containment pressure achieved after the accident (1/2 design pressure in 24 hours) can be maintained. If the water is not replenished, the

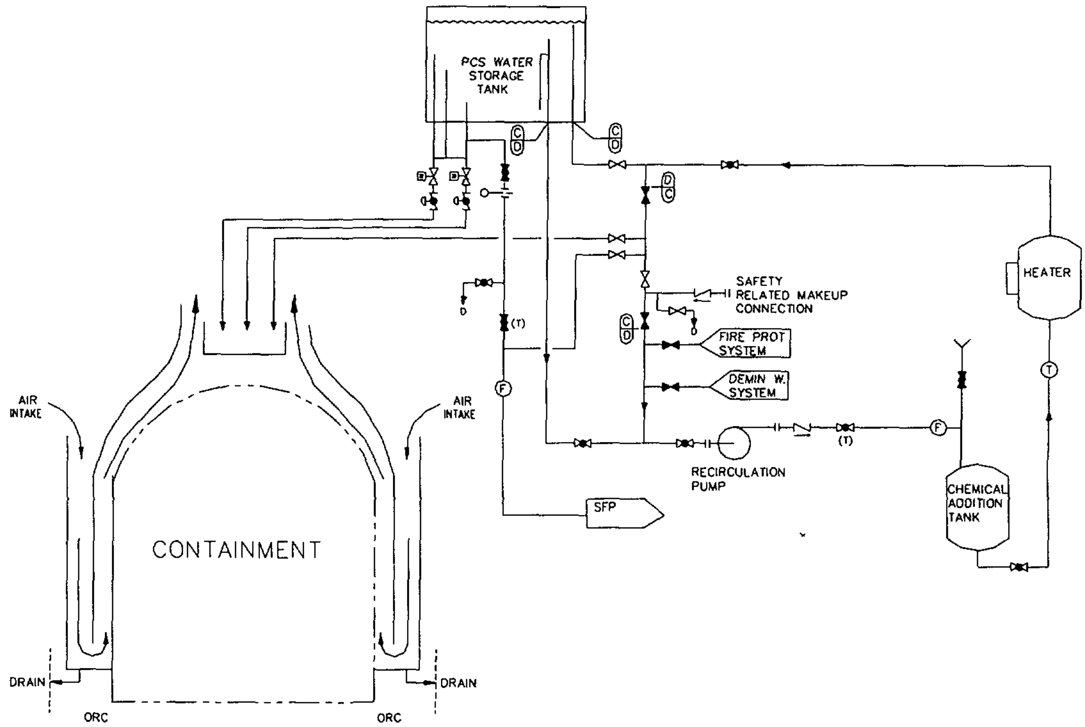

FIG. 7. EP 1000 Passive Containment Cooling System - Simplified Sketch

containment pressure will increase, but the peak is calculated to be well below the Service Level C (as defined by the ASME code) even after two weeks without operator support actions.

The EP 1000 will have a significantly reduced frequency of release of large amounts of radioactivity following a severe accident with core damage scenario. Analysis shows that with only the normal PCS air cooling, the containment pressure will stay well below the predicted failure pressure. Other factors include improved containment isolation and reduced potential for LOCAs outside the containment. This improved containment performance supports the technical basis for simplification of the off-site emergency planning.

3.2 Severe accidents (Beyond Design Basis accidents)

The assessment of the EP 1000 performance against severe accidents is performed in agreement with European Utilities Requirements. The general approach to severe accidents identifies the sequences to be reduced in probability below the credibility threshold and those to be mitigated.

According to the EUR, the assessment of the Design Extension Conditions (DEC) in addition to the Design Basis Accident (DBA) is the preferred method for giving consideration to the complex sequences and severe accidents at the design stage without including them in the Design Basis Conditions.

The assessment of the DEC permits the definition and evaluation of the Design Extension Measures (DEM) to prevent core melting or mitigate the consequences of accident sequences such as:

- Complex sequences which involve failures beyond those considered in the deterministic Design Basis
- Severe accidents, both to prevent early and delayed containment failure and to minimize releases for the remaining conditions that go beyond the Design Basis Conditions (DBC).

The Design Extension concept makes use of probabilistic methods as one way to identify the need for the implementation of measures, including upgraded or additional equipment or accident procedures for complex sequences and severe accidents that provide a significant contribution to the core damage frequency and/or frequency of exceeding acceptable limiting releases.

Several complex sequences and severe accident scenarios, defined in the frame of EP1000 Phase 2A Probabilistic Safety Assessment (PSA) studies, have been analyzed to support Design Extension Conditions (DEC) assessment for the Preliminary Safety Case report.

Two different Severe Accident categories were analyzed, using the MAAP4 computer code in order to:

- assess the external releases following a severe accident with intact containment.
- assess the in-vessel retention capability and analyze the hydrogen behavior inside the containment.

The latter analyses focused in particular on the core melt and relocation time and are used to support the analysis of In-Vessel-Retention (IVR).

The EP 1000 plant relies on the in-vessel retention (IVR) of core debris as the primary severe accident mitigation feature. With the reactor vessel intact and core debris retained in the lower head, there is no need to examine phenomena that may occur as a result of core debris being relocated to the reactor cavity. The EP 1000 reactor vessel insulation is designed to promote water circulation around the reactor vessel external surface. The EP 1000 containment event trees include a node to ascertain whether the reactor coolant system (RCS) is depressurized and a node to determine if sufficient water is available in the cavity to cool, from the outside, the reactor vessel (and debris in the lower plenum).

Success at both of these nodes is required to demonstrate that vessel failure is physically unreasonable.

The engineered design features of the EP 1000 containment promote flooding of the containment cavity region during accidents, and thereby submergence of the reactor vessel lower head. Liquid effluent released through the break during a LOCA event is directed to the reactor cavity. The EP 1000 design also includes provision for draining the in-containment refueling water storage tank (IRWST) water into the reactor cavity through simple operator action. Therefore, the reactor pressure vessel lower head will always be submerged in water in the long term after a LOCA.

Keeping the core debris in the vessel eliminates the need for consideration of ex-vessel events, such as ex-vessel steam explosion and corium concrete interaction.

The main prerequisite to be satisfied to assume IVR are:

- the RCS is depressurized.
- the reactor vessel is submerged above the top of the in vessel debris bed.
- the reflective insulation does not impede water cooling of the vessel.
- the external surface treatments do not impair wetability of the vessel.

Specific IVR capability studies, performed in the frame of Phase 2A activities, have lead to modification of the vessel internals design. The modifications ensure that in the final state of relocation of molten materials in the vessel bottom head, the core support plate and the radial reflector will be included in the molten pool. This significantly reduces the so-called focusing effect due to the molten metallic layer of the pool. However, additional analyses are planned for the next phase of the EPP program to provide confidence of the viability of the IVR strategy for corium cooling. These analyses will mainly focus on the failure mechanisms of the core barrel and corium relocation mechanisms and timing.

3.3 Probabilistic Safety Assessment Highlights

Detailed PSA Level 1 (core damage frequency), and Level 2 (large release frequency) models, for internal events only, have been developed during Phase 2A. The models are consistent with the EP1000 design configuration and operation requirements and the EUR PSA methodology requirements.

The objectives of the EP 1000 Probabilistic Safety Assessment are to:

- Provide an integrated view of the EP 1000 behaviour in response to transients and accidents, including severe accidents.
- Demonstrate that the design meets the proposed safety goals in the EUR document for core damage frequency (CDF) and large fission product releases: that the core damage frequency is less than or equal to 10^{-5} events per reactor year and frequency of a severe release is less than or equal to 10^{-6} events per reactor year for those sequences potentially exceeding "no emergency action" limit.
- Provide input to the design process (that is, provide a tool to investigate detailed design solutions and operational strategies to optimize EP1000 safety).

The PSA results indicate that the EP1000 design largely meets the goals specified above. The CDF and Large Release Frequency (LRF) for at-power internal events (excluding shutdown, fire, and flood events) are ~8.E-08 events per reactor-year and ~1.4E-08 events per reactor-year, respectively. These frequencies are at least two orders of magnitude less than a typical pressurized water reactor plant.

This reduction in risk is due to many improved plant design features, with the dominant reduction coming from highly reliable and redundant passive safety systems. These passive systems are much less dependent on operator action and support systems than current operating plant systems.

The value of predicting very small numbers for probabilities is perhaps in the process rather than in the answer. A well-done PSA that results in a very low number should mean that all sequences that lead to accidents have been quantitatively evaluated and the design and procedures have been progressively modified to reduce the PSA result. This process, if professionally and thoroughly done, and reviewed, should result in an extremely safe operation, even if the predicted probability value is so low as to be uncertain.

The dominant initiating events are:

- Safety injection (SI) line break (35.5 percent)
- Large loss-of-coolant accident (20.5 percent)
- Anticipated transient without scram precursor with loss of main feedwater (17.3 percent)
- Reactor vessel rupture (12.2 percent)
- Steam generator tube rupture (5.4 percent)

The at-power core damage results are dominated (more than 70 percent) by various LOCAs. Thirty-five percent of the contribution is due to the safety injection line break, which is a special initiator, in that its occurrence partially defeats features incorporated into the plant to respond to losses of primary coolant. Even though the safety injection line break core damage frequency dominates the results, its value is very small (one event in 100 million reactor years), with little credit for non-safety systems.

Anticipated Transients Without Scram (ATWS) sequences contribute about 18 percent of the at-power core damage frequency, in part due to modeling simplifications whereby, in the absence of specific modeling and success criteria, it has been assumed that core damage will occur given certain combinations of failures. It is expected that, in the next Phase of the program, the use of specific EP1000 success criteria, and the credit for specific features able to successfully mitigate the consequences of ATWS (e.g., low boron core design), will significantly reduce the ATWS contribution to CDF.

Reactor vessel rupture sequences contribute about 12 percent of the at-power core damage frequency. This is mainly due to a conservative assumption that the failure occurs below the top of the core such that water can not be maintained in the reactor vessel.

The steam generator tube rupture event contributes only about 5.4 percent of the at-power core damage frequency. Compared to operating pressurized water reactors this is a very low contribution. Among the reasons for the small steam generator tube rupture core damage contribution are the following:

- The first line of defense is the startup feedwater system and chemical and volume control system.
- A reliable safety-classified passive residual heat removal system coupled with the core makeup tank sub-system, which provides automatic protection.
- A third line of defense is use of the automatic depressurization system and in-containment refueling water storage tank for accident mitigation should the above-mentioned systems fail.

4 TESTING

The AP-600 validation test programs for AP-600 were utilized as a starting point for EP 1000. The performance of the AP-600 passive safety systems is assessed through analysis of the plant by using validated safety analysis computer codes. The AP-600 test and analysis program has been developed to provide the data and associated analysis to develop and verify the computer code models needed to confidently predict the behaviour of key AP-600 passive safety systems, structures and components. The tests and analyses are integrated so that the AP-600 safety analyses computer codes will meet Nuclear Regulatory Commission (NRC) approval.

Testing has been ongoing since 1988, beginning with the earliest basic research testing under the Advanced Light Water Reactor (ALWR) Technology Programs sponsored by DOE and EPRI. The AP-600 test program evolved into a larger program, with an emphasis on the tests required for Design Certification (DC).

Since the European Passive Plant (EPP) reference design includes most of the same passive safety features and components as the AP-600, most of the testing performed on the AP-600 will be directly applicable to the EPP. However, differences in the number of loops and in the specific passive core cooling system configuration may result in the need for some additional tests for the EPP.

During Phase 2A of the EPP program, the specific set of tests considered necessary to support licensing of the EP 1000 were identified based on the assessments of large LOCA, small LOCA, transient, containment, and long-term cooling thermal-hydraulic processes and differences between the EP 1000 and the AP-600 passive safety system designs. These tests together with the extensive AP-600 tests and earlier PWR plant tests provide sufficient data to validate the safety analysis computer codes that will be used to license the EP 1000.

Because the EP 1000 passive systems are similar to the AP-600 passive systems, the basic research tests performed for the AP-600 are sufficient for the EP 1000. Basic research tests are experimental in nature and are used to provide engineering guidance or detailed information on specific processes to be studied. These tests are also used to determine the feasibility of an engineering concept before proceeding to a larger-scale test or development program.

The only component test considered necessary for the EP 1000 is for the IRWST injection and containment recirculation check valves. This test is necessary because the EP 1000 uses a different type of check valve design, a nozzle type design instead of a swing disk type. This test will determine the opening differential pressure and low flow/part open behaviour of these valves. This test may not be necessary it sufficient vendor data exists.

No separate effects tests are considered necessary for the EP 1000.

The differences between the AP-600 and the EP 1000 passive systems warrant an integral systems test for computer code validation. The interaction between the different passive safety system features is of interest since there are small driving forces that can be affected by small changes to the systems design. Also, there is potential concern about possible active system and passive safety interactions at high pressure.

A full-height, high-pressure integral systems test (like SPES-2) that addresses small LOCA, SGTR, and SLB to examine passive safety system performance and possible active/passive system interaction is considered necessary to support licensing of the EP 1000. Based on the results of AP-600 SPES-2 testing, this test should be able to deal with the interactions of the passive and active systems down to IRWST cut-in pressures. As a result a low-pressure, gravity-driven integral systems test (like OSU) is not considered necessary.

The integral systems test provides data for verifying the component models and correlations developed from component and separate effects tests, thereby validating the reactor systems computer codes.

The specific design of this integral test facility has not been developed although a modified SPES-2 facility should be adequate.

5 PLANT LAYOUT

5.1 Site plan description.

The site plan for a single-unit EP1000 standard plant is shown in Figure 8. The power block complex consists of six principal building structures: the nuclear island, the turbine building, the annex building, the diesel generator building, and the radwaste building. Each of these building structures is constructed on individual basemats. The nuclear island consists of the containment building, and the auxiliary building, all of which are constructed on a common basemat.

5.2 Plant arrangement description.

The EP 1000 plant design helps to minimize the construction schedule and cost, and meets the safety, operational, maintenance, and structural criteria. The most direct means for helping to minimize the construction schedule and the capital cost is to minimize the building volume and the bulk quantities while satisfying all layout criteria, specifically the criteria related to maintenance activities and maintenance space allocation. The EP 1000 arrangement criteria provide a significant increase in maintenance space allocation in many areas as compared with most conventional plants.

The EP 1000 plant arrangement is designed so that operational and maintenance activities can be accomplished in an efficient manner. A key objective is to provide confidence that the utility needs and expectations for operability and maintainability are appropriately included in the design. The plant arrangement contains a mixture of conventional and unique features that facilitate and simplify operational and maintenance activities.

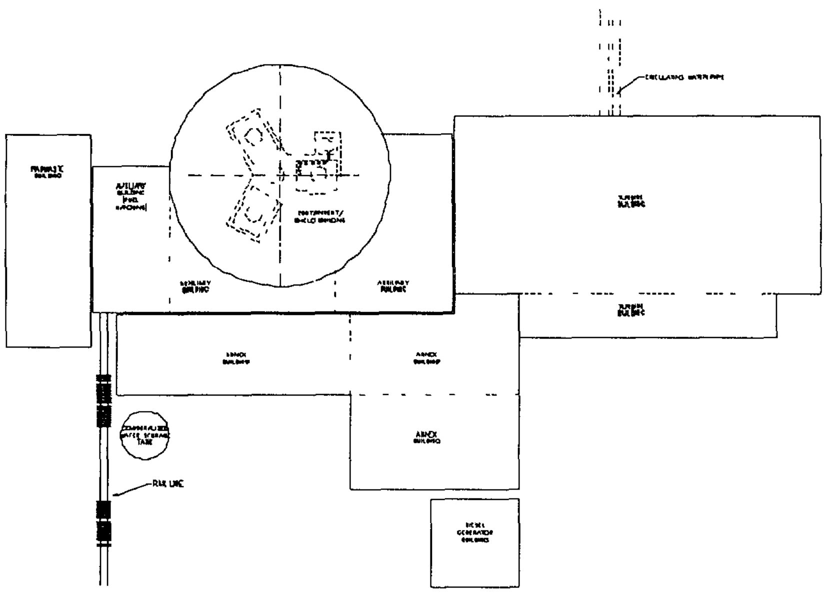

FIG. 8. Site plan for a single-unit EP 1000 standard plant

The EP 1000 plant arrangement provides separation between safety and non-safety equipment, radioactive and non-radioactive equipment, and mechanical and electrical equipment. These separation features in the equipment layout also facilitate maintenance. The radioactive equipment and piping are arranged and shielded to minimize radiation exposure, which is beneficial to the operation and maintenance activities.

Representative general arrangement drawings of the containment building and Nuclear Island are shown in Figures 9 and 10.

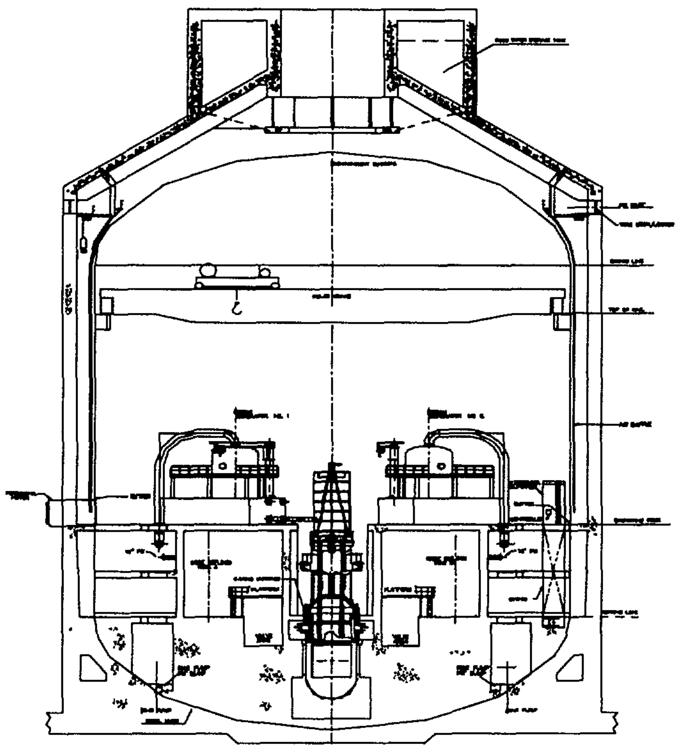

FIG. 9. General containment arrangement; elevation view at Section B-B

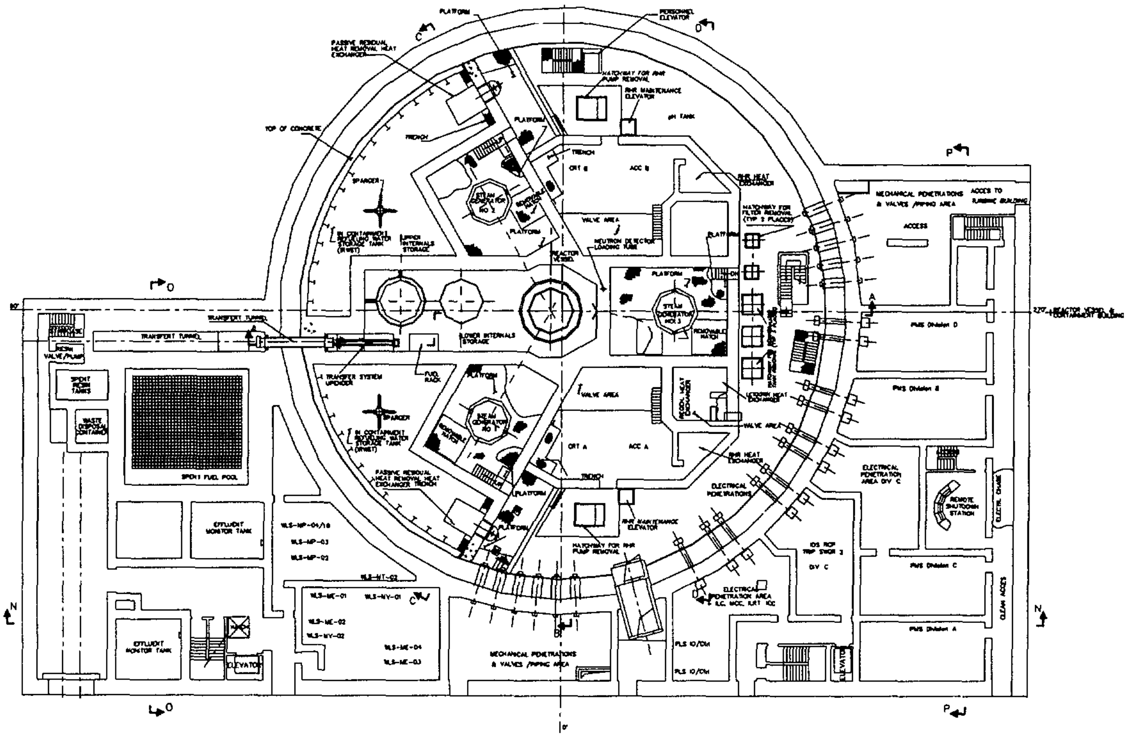

FIG. 10. Nuclear island arrangement; plan view at operating floor elevation

5.2.1 Containment (Reactor Building)

The containment building consists of the containment and all structures within the containment. It is an integral part of the overall containment system with the functions of containing the release of airborne radioactivity both following postulated design basis accidents and Design Extension Conditions. Containment building structures also provide shielding for the reactor core and the reactor coolant system during normal operations.

Two containment designs are considered for the EP 1000 - Single Steel Containment (SSC) and Double Concrete Containment (DCC) [4]. In the following, only the SSC reference configuration will be discussed.

The containment vessel is an integral part of the Passive Containment Cooling System (PCS). A Seismic Category I structure, the containment vessel is a free standing steel cylinder, 46 meters in diameter and 67,6 meters from the containment sump to the inside containment top head. It is surrounded by a Seismic Category I reinforced concrete shield building that provides protection against Aircraft Crash.

There are three floor elevations (grade access, maintenance floor, and operating deck) and ten equipment compartments within the containment building. Floor gratings are provided for access to equipment at other elevations. The principal systems located within the containment building are the Reactor Coolant System (RCS), the Passive Core Cooling System (PXS), the Normal Residual Heat Removal System (RNS), and the Chemical and Volume Control System (CVS).

The shield building roof is conical and the containment bottom shape has been changed from a flat bottom, derived from the SPWR, to an elliptical bottom shape derived from the AP-600 design. This change provides a more robust design and facilitates the licensing process since the US NRC has approved the design for the AP-600.

5.2.2 Auxiliary building.

The primary function of the auxiliary building is to provide protection and separation for the Seismic Category I mechanical and electrical equipment located outside the containment building. It also provides shielding for the radioactive equipment and piping that is housed within the building.

The auxiliary building is a Seismic Category I reinforced concrete structure, which shares a common basemat with the containment building. The auxiliary building is a C-shaped section of the nuclear island that wraps around approximately 70 percent of the circumference of the shield building.

Floor slabs and the structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

The auxiliary building is located between the containment building and the turbine building and between the containment building and the annex buildings. Because of the configuration, the auxiliary building provides communication between the containment and the annex buildings.

6 PROJECT STATUS AND PLANNED SCHEDULE

The ultimate objective of Phase 2 of the program is to develop design details and perform supporting analyses to produce a Safety Case Report for submittal to European Safety Authorities.

The first part of Phase 2, "Phase 2A" is focusing on the definition and design of important systems and structures. Activities have already been performed both to define the design details of the important systems (e.g. Reactor Coolant System, Passive Injection and Core Cooling System, Passive Containment Cooling System, etc.), and to address some specific EUR requirements including Hazards (i.e., Aircraft Crash, Gas Cloud Explosion), Design Extension Conditions and performance requirements (e.g., MOX Fuel, Low Boron Core, etc.) and finally EUR specific Site Interface Requirements (i.e., Seismic Margins, Soil Characteristic and site environmental conditions).

This Phase will be completed at the end of 1998.

In parallel to the Phase 2A effort, a group of European Utilities are sponsoring the activities for the preparation of the EP1000 EUR Volume 3. Volume 3 will be the EP1000 plant example and compliance assessment against the EUR. The EP 1000 EUR Volume 3 program began in June 1997 and will be concluded at the end of 1998.

The evaluation of the EP 1000 design against EUR has shown, to date, only minor non-compliance that are traced and will be solved in the next phase of the EPP program.

The second part of Phase 2, "Phase 2B", will start at the beginning of 1999 and will be completed in the 2001.

Phase 2B includes both the analyses and evaluations required to demonstrate the adequacy of the design, and the preparation of Safety Case Report.

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SWR 1000 — AN ADVANCED BOILING WATER REACTOR WITH PASSIVE SAFETY FEATURES

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Abstract

The SWR 1000, an advanced BWR, is being developed by Siemens under contract from Germany's electric utilities and with the support of European partners. The project is currently in the basic design phase to be concluded in mid-1999 with the release of a site-independent safety report and costing analysis. The development goals for the project encompass competitive costs, use of passive safety systems to further reduce probabilities of occurrence of severe accidents, assured control of accidents so no emergency response actions for evacuation of the local population are needed, simplification of plant systems based on operator experience, and planning and design based on German codes, standards and specifications put forward by the Franco-German Reactor Safety Commission for future nuclear power plants equipped with PWRs, as well as IAEA specifications and the European Utility Requirements. These goals led to a plant concept with a low power density core, with large water inventories stored above the core inside the reactor pressure vessel, in the pressure suppression pool, and in other locations. All accident situations arising from power operation can be controlled by passive safety features without rise in core temperature and with a grace period of more than three days. In addition, postulated core melt is controlled by passive equipment. All new passive systems have been successfully tested for function and performance using large-scale components in experimental testing facilities at PSI in Switzerland and at the Jülich Research Centre in Germany. In addition to improvements of the safety systems, the plant's operating systems have been simplified based on operating experience. The design's safety concept, simplified operating systems and 48 months construction time yield favourable plant construction costs. The level of concept maturity required to begin offering the SWR 1000 on the power generation market is anticipated to be reached, as planned in the year 2000.

1 INTRODUCTION

The SWR 1000, an advanced boiling water reactor concept, is being developed by Siemens under contract from and in close with Germany's electric utilities and with the support of European partners in Finland (TVO), the Netherlands (KEMA), Switzerland (PSI) and Italy (ENEL). This development project is currently in its Basic Design Phase, which will be concluded in mid-1999 with the release of a site-independent safety report and costing analysis of projected erection costs.

In parallel with the design phase, an experimental testing program is being conducted at Siemens' own testing facilities and at other German and European research centres to provide verification of the function and effectiveness of the SWR 1000's passive safety systems.

2 NUCLEAR STEAM SUPPLY SYSTEM

The nuclear steam supply system is located in the reactor building and is surrounded by a steel-reinforced concrete containment equipped with an inside liner. Table 1 presents the key technical data for the SWR 1000 in comparison with those of a traditional 1300 MW BWR plant.

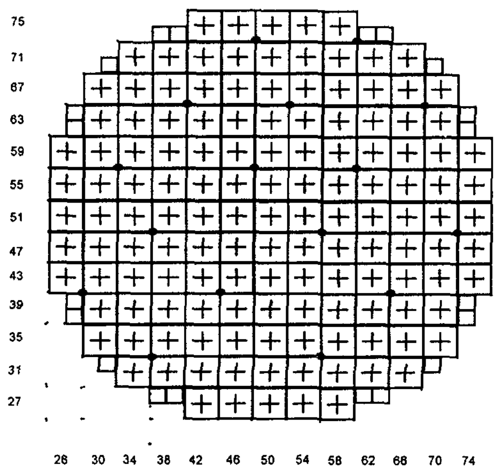
2.1 Reactor Core and Fuel Assemblies

The SWR 1000 core (Figure 1) represents an evolutionary further development of previous standard BWR core designs. While no changes have been made to the basic structure of the BWR

Table 1: Key technical data of SWR 1000 compared to an advanced 1300 MW BWR plant

Data		SWR 1000	1300 MW BWR*
Overall plant			
- thermal output	MW	2778	3840
- gross electric output	MW	1013	1428
- net electric output	MW	977	1373
- net efficiency	%	35 2	35 7
Reactor core			
- No of fuel assemblies	-	568 (13x13)	784 (10x10)
- Total uranium weight	t	121	138,1
- Active height of core	m	2 80	3 71
- Average power density	kW/l	47	56 8
- Average discharge burnup	GWd/t	65	50
- Average enrichment	wt %	5 45	3 63
- Core throughput	kg/s	12000	14300
Reactor pressure vessel			
- Inside height	m	22 55	22 35
- Inside diameter	m	7 0	6 62
- Design pressure	bar	88	87 3
- No of recirculation pumps	-	6	8
Turbine			
- Number	-	1	1
- Speed	1/min	3000	1500
- No of HP/LP casings	-	1/3	1/2
Containment			
- Inside diameter	m	32 0	29
- Inside height	m	28 7	32 5
- Design pressure (abs)	bar	7 5	5 3
- Drywell volume + gas volume of core flooding pool	m^3	5700	8200
- Water volume of pressure suppression pool	m^3	2900	3100
- Gas volume of pressure suppression pool	m^3	5500	6000
- Water volume of core flooding pool	m^3	3100	-
Plant design life	years	60	40
Plant construction period	months	48	60

* Siemens Gundremmingen Nuclear Power Station, Unit B+C



Number of fuel assemblies	*568*
Number of control rods	137
Control rod pitch	387 mm
Core diameter	5500 mm
Local power range monitor assemblies	16

FIG 1 Arrangement of SWR 1000 Reactor Core

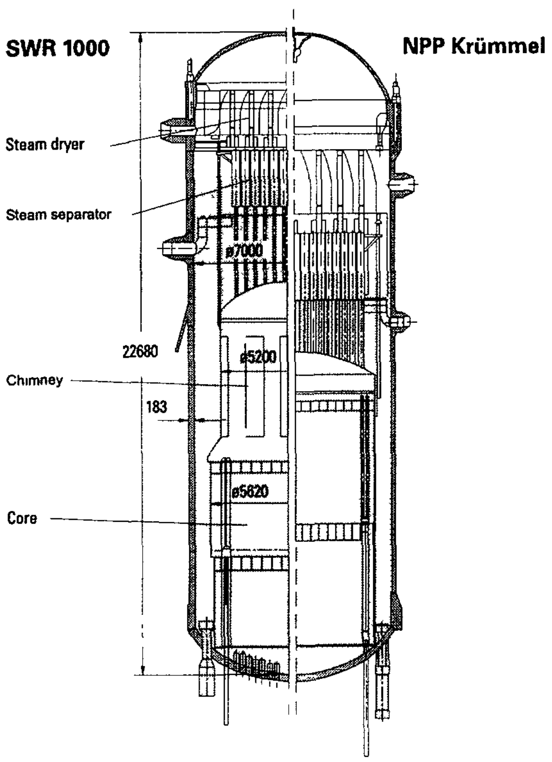


FIG. 2: Reactor Pressure Vessel

core design, certain modifications have been introduced. These adaptations include reducing the active height of the core and increasing the size of the fuel assemblies.

By reducing the active core height, the core can be positioned lower down inside the RPV. As a result, there is a greater water inventory available inside the RPV above the core, which facilitates accident control.

The aforementioned modification of the fuel assemblies consisted of enlarging the existing ATRIUM™10 fuel assembly design (10x10-9Q) to a 13x13-25Q (ATRIUM™13 fuel assembly) rod configuration. Fuel rod diameter and pitch, on the other hand, remain unchanged from the ATRIUM™10 fuel assembly. As a result of this new design, there are fewer fuel assemblies in the core, which reduces handling times during refueling. Reducing the number of fuel assemblies also lowers the number of control rods required, and hence the number of control rod drives as well. The number of in-core instrumentation assemblies and power distribution detectors can also be reduced.

The average power density of the core has been reduced from 56,8 to 47 kW/l. This reduction, together with the advanced fuel assembly design, ensures good plant behavior during transients.

Flexible operating cycles are planned for the SWR 1000. For example, the core can be operated in annual cycles or in cycles lasting anywhere from 1 to 2 years. The desired mean discharge burnup is 65 GWd/t.

All of these core design modifications contribute to the economic efficiency of SWR 1000 operation.

2.2 Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) (see Figure 2) encloses the reactor core and the RPV internals. Its main dimensions are comparable to those of the RPV in a Siemens 1300 MW BWR plant (see Table 1), which means that the SWR 1000 RPV is of larger volume in relation to actual core power. As a result, and due to the low positioning of the core inside the RPV, core uncovery is prevented in the event of automatic depressurization, even without coolant makeup during pressure drop.

The core shroud, as well as the top upper and lower core grid, mainly serves to align the core, control rods and core instrumentation and to guide core flow. Steam separators and steam dryers are installed in the RPV to separate the steam-water mixture leaving the core. A chimney is located between the core and the steam separators.

All RPV internals are designed to allow removal and replacement as needed.

The RPV is supported against the building via a frame mounted around the RPV top half. The RPV internals, such as the core shroud, upper and lower core grid, steam separators and steam dryers, for example, are essentially based on the proven technology used in Siemens 1300 MW BWR design.

2.3 Reactor Coolant Pumps

The reactor coolant pumps provide flow of coolant through the core. Comparative studies of natural and forced coolant circulation in the RPV have shown that it is advantageous to retain the forced circulation flow provided by internal reactor coolant pumps in the SWR 1000 design owing to the benefits gained in terms of fuel utilization and load cycling capability. The design calls for six reactor coolant pumps.

Unlike the previous standard pump design used in German BWR plants, a wet-rotor pump is planned for the SWR 1000, which requires neither mechanical seal nor oil supply. This design offers certain operational advantages, and has been proven in the Swedish and Finnish BWR plants from ABB Atom.

2.4 Control Rod Drives

The SWR 1000 retains the control rod drive design proven by operating experience at existing BWR plants. However, plans call for the long drive components (hydraulic drive unit and threaded spindle) in future to be installed and removed from above through the RPV. Only the electric motor drive unit and the seal housing will be installed and removed from the control rod drive compartment below the RPV as before. This modification enables the required withdrawal height in the control rod drive compartment - and hence the building height - to be reduced by the equivalent of the drive length.

2.5 Fuel Assembly Handling and Storage

There is no significant difference from a function point of view between the equipment and plant structures used for refuelling, storage of new and spent fuel assemblies and handling of reactor components in the SWR 1000 and those equipment and plant structures found in standard BWR nuclear power plants.

2.5.1 Store for New Fuel Assemblies

New fuel assemblies are stored in the new fuel store specially provided for this purpose, which is located adjacent to the spent fuel pool. The new fuel assemblies are placed in dry storage in racks that can accommodate some 270 fuel assemblies.

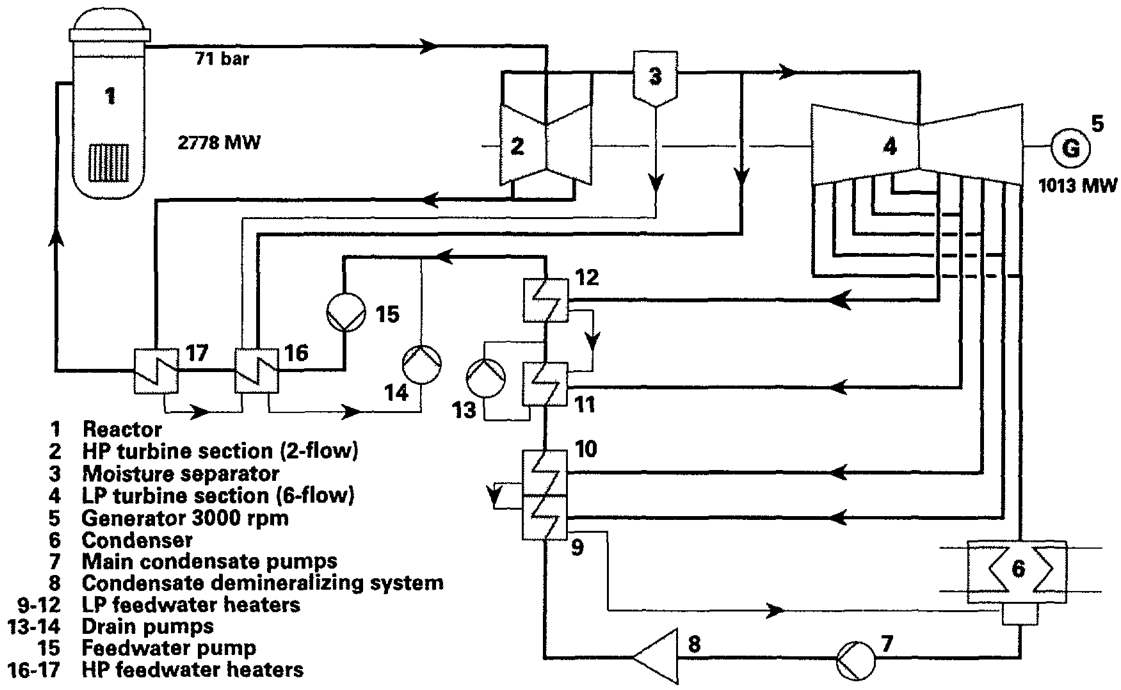


FIG. 3: Steam, Condensate and Feedwater Cycle

2.5.2 *Spent Fuel Pool*

Spent fuel assemblies are stored in the water-filled fuel pool located in the reactor building on an extension of the axis of the dryer-separator storage pool and reactor well. The pool water ensures residual heat removal and provides shielding.

Owing to their reduced length, the fuel assemblies are stored in two layers in high-density storage racks with inserted boron channels to maintain subcriticality. Racks for storage of control rods and RPV internals (such as in-core instrumentation assemblies, neutron sources, etc.) are provided in addition to the fuel assembly storage racks. Total storage capacity allows accommodation of some 1400 fuel assemblies and approximately 170 control rods.

3 STEAM, CONDENSATE AND FEEDWATER CYCLE

Like the boiling water reactors in operation today, the SWR 1000 operates according to the direct-cycle principle (Figure 3), i.e. the live steam generated in the RPV passes directly to the double-flow high-pressure (HP) section of the steam turbine via three (instead of the previous four) main steam lines fitted with combined stop and control valves. After undergoing partial expansion in the HP turbine section, the steam is passed through a moisture separator to the three double-flow low-pressure (LP) sections of the turbine. Reheating upstream of the low-pressure turbine sections - a feature of previous designs - is eliminated in the SWR 1000 as, thanks to continuous development in the turbine sector (including in particular dewatering capabilities in the individual turbine stages), the increase in efficiency achieved today by reheating is insignificant. This translates into savings in investment and maintenance costs.

The condensate is removed from the condensers of the three LP turbine sections and returned to the RPV at a temperature of 210 °C via a single-train (as opposed to the previous double-train) feed

Pos	Number
1 Wetwell	1
2 Condensation pipe	16
3 Spring loaded pilot valve	8
4 S&R valve	6
5 Scram system	2
6 Core flooding pool	4
7 Emergency condenser	4
8 Passive pressure puls transmitter	3 x 4
9 Pilot valve	8
10 Core flooding system	4
11 Dryer-separator storage pool	1
12 Containment cooling condenser	4
13 Passive outflow reducer	4

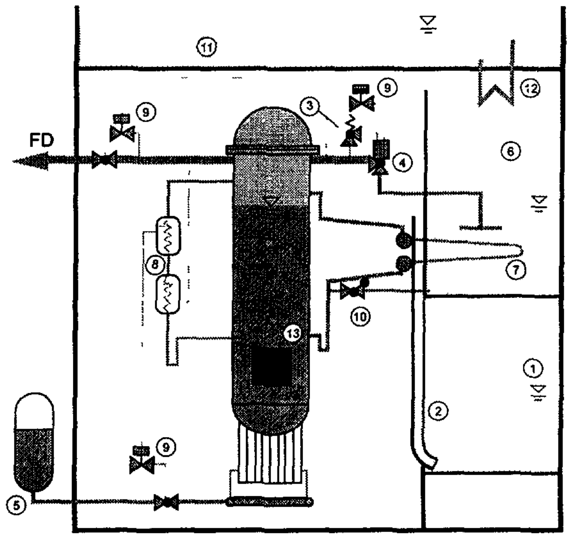


FIG. 4: Passive Shutdown, Core Flooding and Residual Heat Removal Systems

heating system comprising a condensate demineralizing system, LP feedwater heater, HP feedwater heater and two pump units. The LP feedwater heater units 1 and 2 are designed in a duplex arrangement and installed horizontally in the condenser neck. Two combined pump units will be used for the condensate and feedwater pumps, comprising motor, condensate pump as re-entry pump and feedwater pump. In the event of loss of one pump unit, the other unit delivers 75% of rated flow. Feedwater injection is via two feedwater lines (four in previous BWR designs) connected to the RPV. In the SWR 1000, the feedwater tank is eliminated and replaced by a surface feedwater heater. This concept is supported by the excellent operating experience gained from existing BWR plants.

The turbine generator set consists of a single-shaft, saturated steam turbine coupled directly to a three-phase AC synchronous generator. The planned 3000 rpm design costs less and is smaller than a 1500 rpm unit, which also allows the turbine building to be of smaller dimensions.

4 PASSIVE AND ACTIVE ACCIDENT CONTROL SYSTEMS

4.1 Overview

The primary objective in developing the SWR 1000 is to enhance the quality of safety by introducing passive systems (Figure 4) for performing safety-related functions in the event of transients or accidents. Compared to the reactors of today, the technology employed in these systems is much simpler, operation of the equipment being independent of a power supply and activation by I & C systems.

Passive systems are characterized by the fact that they utilize the laws of nature (e.g. gravity, pressure differentials) to perform their designated safety functions and dispense with active components (e.g. electromotor driven valves and pumps).

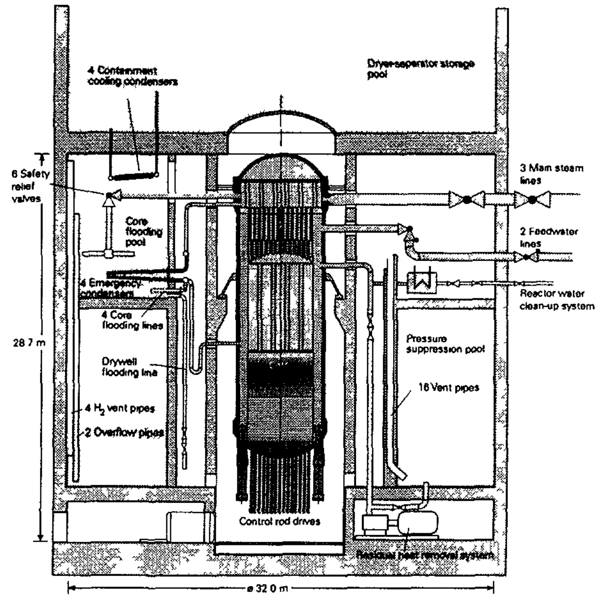


FIG 5 Containment and Internals

4.2 Containment and Passive Safety Features

4 2.1 Containment

The primary function of the containment is to protect the environment against any release of radioactive materials or line-of-sight radiation under all accident conditions.

Like all other recent-generation BWR plants, a cylindrical containment made from steel-reinforced concrete equipped with an inner liner and pressure suppression system was selected (Figure 5). The containment is divided into a drywell and a pressure suppression pool, as required by the pressure suppression system.

The containment design pressure also takes account of the hydrogen release arising from a postulated 100% oxidation of the zirconium present in the RPV in the event of a core melt accident.

4 2.1.1 Drywell

In addition to the RPV and the three main steam lines and two feedwater lines, the following components are located in the drywell: four large hydraulically-linked core flooding pools, the emergency condensers and containment cooling condensers for passive heat removal, the flooding lines for passive flooding of the RPV and the passive pressure pulse transmitters for initiation of safety functions without the need for actuation by I&C systems. In addition, the drywell is equipped with two 100 % capacity recirculation air cooling systems. The high-pressure zone of the reactor water cleanup system (HP cooler and pressure-reducing station) and the lines of the residual heat removal system are also located inside the drywell. Thanks to the shorter control rod drives and a design which allows the long control rod drive components to be removed from above, the RPV can be positioned lower down inside the containment.

The main steam lines and feedwater lines connected to the RPV are each equipped with two isolation valves, one located inside and one outside the dedicated containment penetrations. Apart

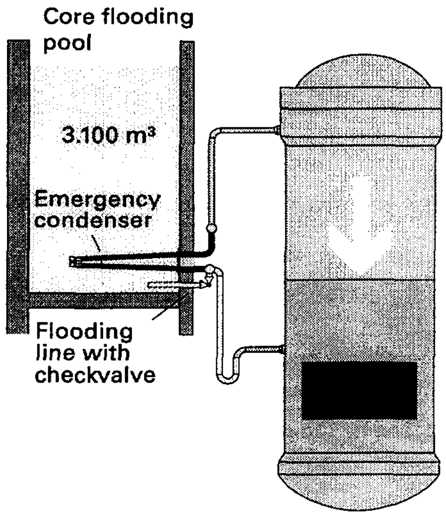


FIG. 6: Core Flooding by Gravity Flow

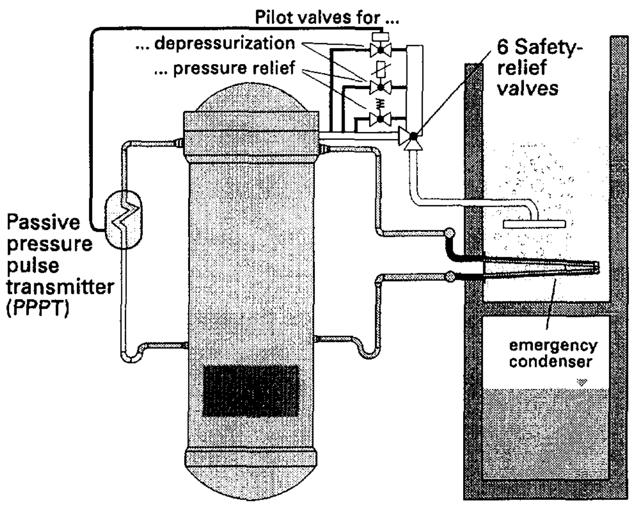


FIG. 7: RPV Safety Relief Valve System

from the main steam and feedwater lines, there is no high-energy piping conveying reactor water (with the exception of instrumentation lines) exiting the containment; the isolation valves of these remain open during operation.

The entire containment is inerted with nitrogen during normal operation to reliably prevent hydrogen-oxygen reactions which can result from a serious core melt accident, thereby ensuring fire protection during operation.

4.2.1.2 Pressure Suppression Pool

The pressure suppression pool performs the following tasks:

- acts as a heat sink in the event of accident conditions
- provides a water inventory for active RPV makeup via the LPCI-residual heat removal system.

As part of the pressure suppression system, the pressure suppression pool is located between the outer and inner cylinder below the core flooding pools and is one-third filled with water. The pressure suppression pool is connected to the drywell via vent pipes concrete-embedded into the inner cylinder.

4.2.1.3 Core Flooding Pools

The core flooding pools act as a heat sink for the emergency condensers and the safety relief valve system. In addition, owing to the pool elevation, the water in the core flooding pools is used for passive flooding of the reactor core following RPV depressurization in the event of a LOCA. In this function, spring check valves open the flooding lines automatically (Figure 6). Passive flooding serves as a diverse supplementary function to the active injection systems for core cooling.

In the event of a serious core melt accident, the water inventory in the core flooding pools is used for cooling the RPV from the outside.

4.2.1.4 Safety Relief Valve System

The tasks of the safety relief valve system are as follows:

- Protection of the reactor coolant pressure boundary against pressure in excess of allowable limits (pressure relief)
- Automatic depressurization of the RPV in the event that the RPV level falls below specified values or in the event of a pressure rise in the containment (LOCA in the containment)
- Short-term removal of excess steam in the event of turbine trip and load shedding
- Open position of valves under depressurized condition to prevent HP core melt path

The safety relief valve system is located inside the containment and consists of the safety relief valves and relief lines with steam quenchers, which are installed in the core flooding pool (Figure 7). This system is thus based largely on the previous, proven system concept used in German BWR plants to date.

4.2.1.5 Emergency Condensers

The emergency condensers function as completely passive devices for residual heat removal from the RPV to the core flooding pool. As a result, the need for HP injection systems is eliminated. The emergency condensers also function in part as a diverse means of depressurization to the safety relief valves down to a pressure which ensures active low pressure coolant injection.

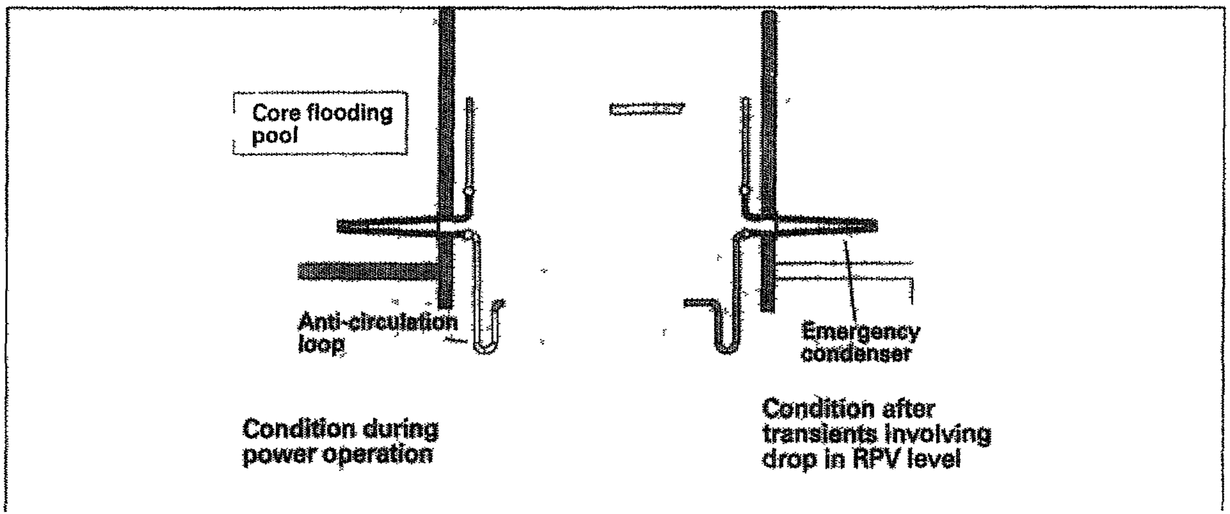


FIG. 8· Emergency Condenser

The emergency condenser system (Figure 8) consists of four separated sub-systems. Each emergency condenser system consists of a steam line leading from an RPV nozzle, and a condensate return line back to the RPV, which is equipped with a siphon. The emergency condensers are connected to the RPV with no isolating element, and are activated only by a drop in the RPV level. The emergency condenser is not in operation during plant operation, as the heat exchanger tubes are filled with cold condensate owing to their location in relation to the normal RPV level, such that heat transfer is not possible. The heat exchanger tubes only fill up with steam (which condenses) after the water level in the reactor falls. The condensate returns to the RPV by gravity flow.

These components were successfully tested at the emergency condenser test facility at Germany's Jülich Research Center using large-scale components, thereby verifying their functional capability and capacity.

4.2.1.6 Containment Cooling Condensers

The containment cooling condensers (CCCs) are designed to remove residual heat passively from the containment to the dryer-separator storage pool located above the containment inside the reactor building following loss of the active residual heat removal systems. The CCCs are actuated by rising temperatures in the containment at increasing steam partial pressure.

The system (Figure 9) consists of four CCCs. The condenser is connected to the dryer-separator storage pool outside the containment via a feed line and a discharge line. The feed line and discharge line and the condenser tubes are filled with water from the dryer-separator storage pool. As the system functions entirely passively, no switching operations are necessary for startup.

In addition to the heat exchangers with feed and discharge lines, the system also includes four H_2 vent pipes lines. The H_2 vent pipe intake is positioned above the heat exchanger and its discharge end is at a submerged position in the pressure suppression pool (Figure 5). They are submerged at a lesser depth than the vent pipes.

The function of these new components was successfully verified at the PANDA test facility at the Paul Scherrer Institute in Würenlingen (Switzerland) using large-scale components.

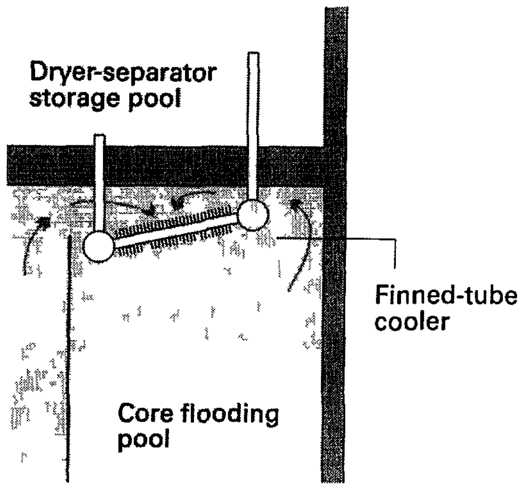


FIG 9 Containment Cooling Condenser

4 2 1 7 Passive Pressure Pulse Transmitter

Passive pressure pulse transmitters (PPPTs) are installed in the SWR 1000 as new, passive switching devices for safety-related switching operatings. The PPPTs, which function without need of electric power supply, external media or actuation via I&C signals, serve to initiate reactor scram, containment isolation of main steam lines, and automatic depressurization of the RPV using system-fluid-actuated valves and valves with stored actuation energy. The PPPTs only commence functioning given a drop in the RPV level

The PPPT (Figure 10) consists of a small heat exchanger, which is connected to the reactor via a non-isolatable pipe. At normal fill level in the reactor, the primary side of the heat exchanger is filled with water such that no heat transfer takes place. When the reactor fill level drops, however, the primary side is emptied and fills with steam, which condenses The water stored on the secondary side

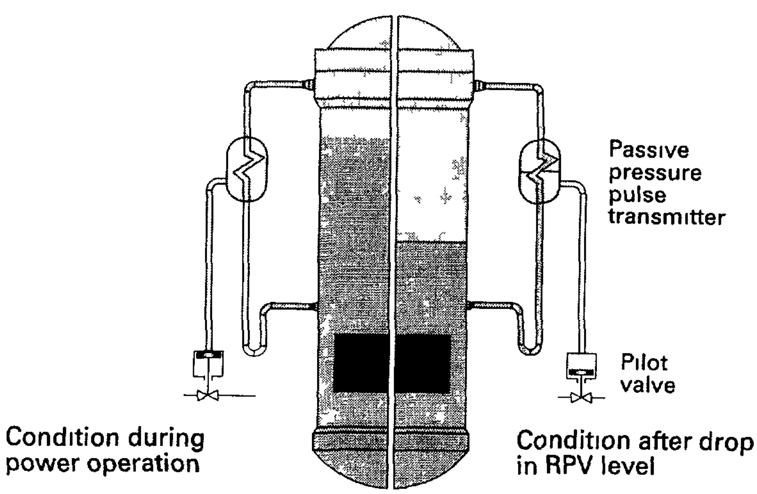


FIG 10 Passive Pressure Pulse Transmitter

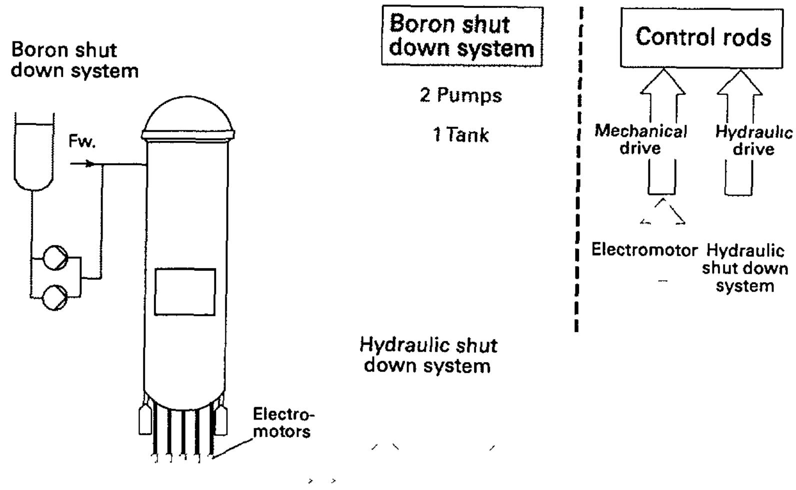


FIG. 11: Shutdown Systems

is thereby heated and partially evaporates, leading to a rapid pressure rise. The pressure rise triggers the safety functions automatically and passively via diaphragm pilot valves.

These devices function entirely independently, and therefore on a diverse basis to the safety I&C equipment. Their integration into the plant's systems engineering is planned (two 2-out-of-2 configuration) such that spurious actuation of a PPPT does not lead to initiation of actions, but also such that loss of a PPPT cannot prevent initiation.

As a new device, the functional capability of the PPPT was tested in the emergency condenser test facility at Germany's Jülich Research Center.

4.3 Reactor Shutdown Systems

Diverse systems are available for shutdown of the reactor (Figure 11). These include the control rods, with their diverse drive systems (electric motor drive for operational shutdown processes, and hydraulic drive for reactor scram). The hydraulic drives are supplied by a scram system. The SWR 1000 is equipped with a boron shutdown system, which functions as a diverse means of reactor shutdown with respect to the control rods and is completely independent of control rod operation and effect.

The concept of the scram system is based largely on the collector tank concept implemented in German BWR plants, whereby the energy required for fast control rod insertion by hydraulic means has to date been stored in tanks under nitrogen pressure. With regard to scram tank pressure medium, a change from nitrogen to steam pressure is currently being investigated. In the latter case, the water-filled tanks used for supplying the hydraulic medium are under a permanent steam pressure blanket generated by electric heating of the top water area. This modification would enable reduction of the tank size and design pressure, and prevent nitrogen from entering the RPV in the event of malfunction of the tank isolation valve.

The planned boron shutdown system is also based on the known concept implemented in all BWR plants: the quantity of pentaborate solution required for hot and cold sub-criticality is stored in an open tank and conveyed to the RPV upon challenge by means of high-pressure pumps. The recirculating pumps of the reactor water cleanup system are used as the high-pressure pumps.

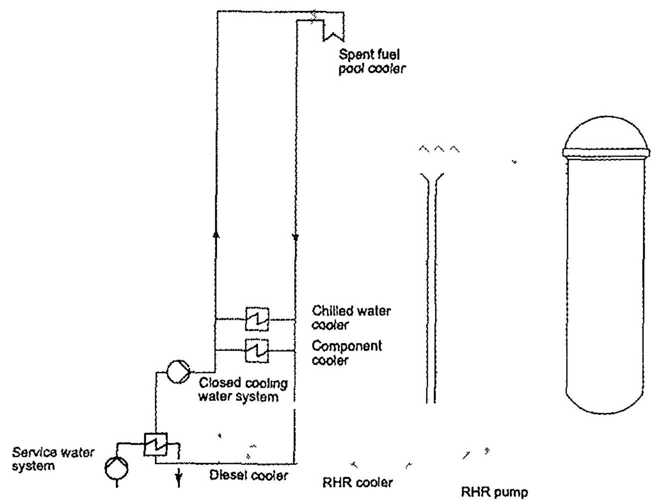


FIG. 12: Active Core Flooding and Residual Heat Removal Systems

4.4 Containment Isolation of Main Steam Lines

The main steam lines are equipped with isolation valves positioned inside and outside the containment at the containment penetrations. The system-fluid-actuated valves are of diverse design, the valve inside the containment being of quick-closing gate-type design, while the external valve is a quick-closing angle valve. Containment isolation is initiated, as in existing BWR plants, via safety I&C systems. In addition, however, a further passive actuation is planned for the SWR 1000, via diaphragm pilot valves arranged in parallel, actuated by PPPT.

4.5 Active Core Flooding and Residual Heat Removal Systems

Due to the additional passive residual heat removal systems, the SWR 1000 concept will include only two active low-pressure core flooding and residual heat removal systems, which are comparable to the systems in existing BWR plants in terms of their range of tasks (Figure 12).

These systems perform the following tasks, as in plants of earlier design:

- Reactor cooling during operational shutdown and in the shutdown condition
- Water transfer operations prior to and subsequent to refuelling
- Operational heat removal from the core flooding pool and pressure suppression pool water
- Heat removal from the containment in the event of loss of the main heat sink by cooling the pressure suppression pool and core flooding pool water
- Low-pressure feed of coolant to the RPV and simultaneous heat removal in the event of loss-of-coolant accidents.

The systems are actuated via safety I&C systems, and system-associated electrical loads are connected to the emergency power supply system.

High-pressure injection systems for the RPV are no longer required in the SWR 1000 design thanks to installation of the emergency condensers.

4.6 Systems for Control of Severe Core Melt Accidents

Loss of all active and passive injection functions and all emergency condensers is assumed for the postulated severe core melt accident. To control this serious accident scenario the following additional safety systems are planned for the SWR 1000 and the plant is designed to withstand the consequences of the accident:

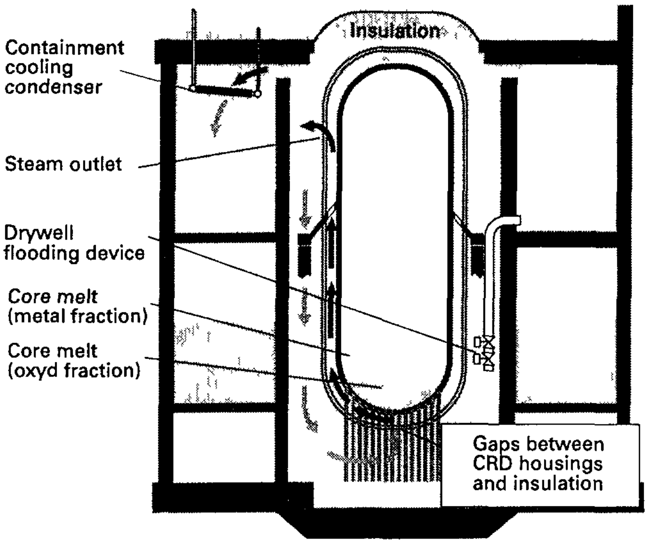


FIG. 13 Cooling of RPV Exterior in the Event of Core Melt

Core melt at high pressure is ruled out by the design of the depressurization system. The core melt is retained in the RPV at low pressure owing to cooling of the RPV exterior. A flooding system is installed for this purpose which feeds into the lower area of the drywell from the gravity core flooding pool (Figure 13). The flooding system is permanently isolated and activated upon challenge. The steam arising from cooling of the RPV from the outside is condensed at the CCCs, which transfer the heat from the containment to the water of the dryer-separator storage pool. Refilling of the dryer-separator storage pool, which only becomes necessary several days after the onset of accident conditions, enables virtually unlimited heat removal.

The containment design is based on the pressure buildup due to the hydrogen arising from a 100 % zirconium-water reaction involving the zirconium inventory present in the core. Hydrogen release always occurs via the drywell, and hydrogen is also partly flushed into the pressure suppression pool depending on the given pressure conditions (cf. Section 4.4.2.4). Any further pressure buildup due to chemical reactions of the hydrogen is not possible, as the containment is inerted with nitrogen.

Long-term pressure relief in the containment after the onset of accident conditions is effected via the off-gas venting system already installed in all current BWR plants, with catalytic hydrogen recombination and the connected holdup system.

5 OTHER REACTOR AUXILIARY SYSTEMS

5.1 Fuel Pool Cooling System

Two redundant cooler units, each consisting of four heat exchangers operating in parallel, are installed in the fuel pool in the SWR 1000 (Figure 12). The fuel pool water is cooled by natural convection. Redundancy is ensured by connecting the cooler units to the redundant closed cooling water systems that are backed up by emergency power supply. The coolers are arranged on the fuel pool wall in such a manner that defined water flows are obtained.

5.1.1 Reactor Water and Fuel Pool Cleanup System

In the SWR 1000, unlike in previous BWR plants, the low-pressure concept is applied (standard practice in pressurized water reactor plants) for reactor water cleanup system. In the SWR 1000

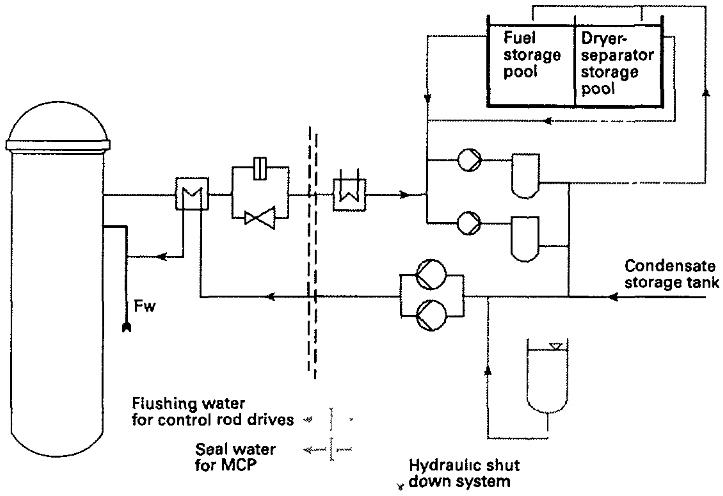


FIG. 14: Reactor Water and Fuel Pool Cleanup System

application, a regenerative heat exchanger and a pressure reducing station are arranged inside the containment, while a further cooler and the powdered resin precoat filter are located outside the containment (Figure 14). The cleaned water is conveyed back into the containment by means of two recirculating pumps and fed into the RPV via the regenerative heat exchangers. The two recirculating pumps also convey the control rod drive flushing water and the reactor coolant pump sealing water. In addition, they are used for the boron shutdown system and as booster pumps for filling the scram tanks. In comparison to high-pressure cleanup, this effective combination of tasks compensates for the disadvantage in energy terms of low-pressure cleanup with pressure reduction and high-pressure pumps.

The advantage of this concept lies in the fact that the filters, with their large number of connections, and the low-pressure cooler can be positioned outside the containment, and the number of containment penetrations can thus be significantly reduced. In addition, the filters, now located in the reactor building, can also be used for cleanup of the fuel pool water and the water in the dryer-separator storage pool.

6 ELECTRICAL AND INSTRUMENTATION & CONTROL SYSTEMS CONCEPT

6.1 Electrical Systems

The passive safety systems are capable of controlling all postulated accidents during power operation. This enables redundancy of active safety systems to be limited to two 100%-capacity trains. As a result, electric power supply to the plant itself (both the auxiliary and emergency power supply grids) are designed on a two-train basis.

6.1.1 Connection to Public Grid

The structure of the electric power supply system is shown in Figure 15. The generator feeds into the public grid via a generator transformer and the main grid connection. The power required for the auxiliary power system is tapped off between the generator and the generator transformer and fed to the 10 kV switchgear via two auxiliary power transformers.

The 10 kV auxiliary power busbars, in addition to being connected to the auxiliary power transformers, are connected to the grid via a backup grid transformer and a backup grid connection. In the event of unit failure, or loss of the main grid connection, power supply for auxiliary power can be guaranteed by switchover to the backup grid connection.

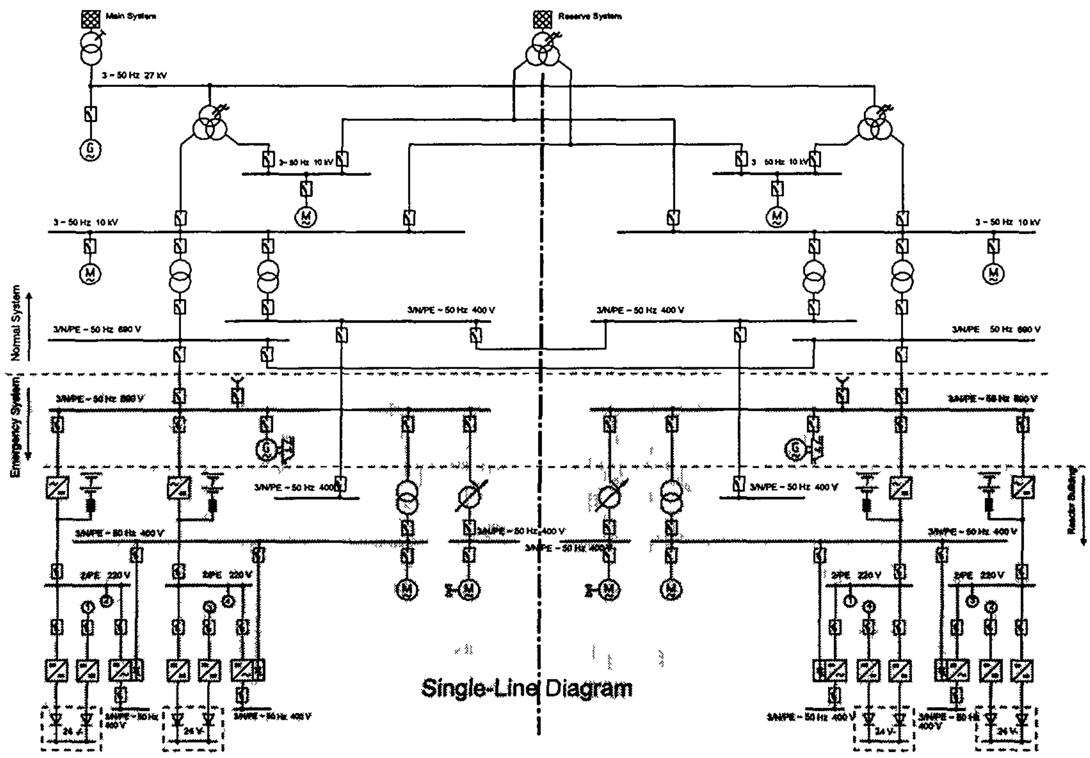


FIG. 15· Electric Power Supply Systems (Single Line Diagram)

6.1 2 Plant Auxiliary Power Grid

In view of the different power levels of the various loads connected to the plant auxiliary power grid, three voltage levels are made available.

For reasons of maintaining voltage stability, only electrical loads with a capacity of < 1 MW are connected to the 10 kV busbars to which the emergency power supply trains are connected. Electrical loads with a higher capacity are connected to the second 10 kV busbar in each train.

One 690 V emergency power supply busbar per train is supplied from the 690 V busbar of the auxiliary power supply grid.

6 1 3 Emergency Power Supply Grid

All electrical loads that have to remain in operation or come on-line in the event of loss of the auxiliary power supply grid are supplied by the emergency power supply grid.

In the event of loss of the auxiliary power supply grid, an emergency diesel generator takes over independent power supply of all connected electrical loads. Electrical loads for which a period without power is allowable during run-up of the emergency diesel generator are connected to the three-phase distributors (690 V and downstream 400 V distribution) of the emergency power supply grid.

A separate busbar is provided for each train for power supply to valve actuators. This busbar is connected to the 690 V busbar via a regulating transformer. This ensures that the valve actuators can be operated reliably and without wear or damage even in the event of major voltage dips in the auxiliary power system.

6 1.4 Uninterruptıble Power Supply

Electrical loads that must remain in operation on an uninterruptible basis or have to be connected immediately in the event of loss of the auxiliary power supply grid are connected to the

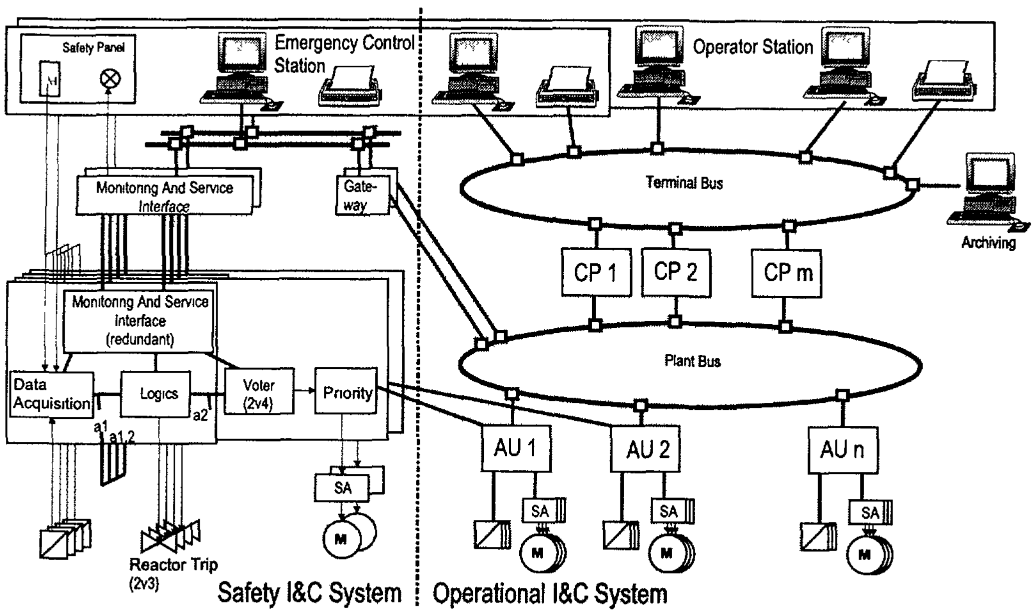


FIG. 16: Instrumentation & Control Concept

uninterruptible power supply. These loads are supplied with power either by the 220 V DC system or via the downstream inverter and the 400 V distributor connected to it, or supplied directly via local uninterruptible power supply systems.

The 220 V DC systems are supplied with power via rectifiers by the 690 V emergency power supply distributors on an individual train basis, whereby two DC systems are provided for each train. Each 220 V DC system has a battery and a charging rectifier.

6.1.5 Power Supply to Instrumentation & Control Systems

The instrumentation and control (I&C) systems are supplied with power at a constant voltage of $\pm$ 24 V by the 220 V DC systems via DC/DC converters. The power is supplied on a double diode-decoupled basis by one 220-V DC system of the dedicated train and one 220 V DC system of the second independent train.

Power is supplied to the monitoring systems and the computers at the master control console in the main control room and in the emergency control station via inverters or uninterruptible power supply systems.

6.2 Instrumentation & Control Systems

Thanks to improvements in the quality of safety in the SWR 1000, achieved through the introduction of additional passive systems to perform safety functions in the event of transients and accidents, the I&C concept can be considerably simplified in comparison with current BWR plants, as the passive safety systems function independently of electrical power supply and without actuation by I&C systems.

The digital I&C concept planned for the SWR 1000 is made up of the following subsystems (Figure 16):

- Operational I&C system
- Safety I&C system
- Process information system (PRISCA)

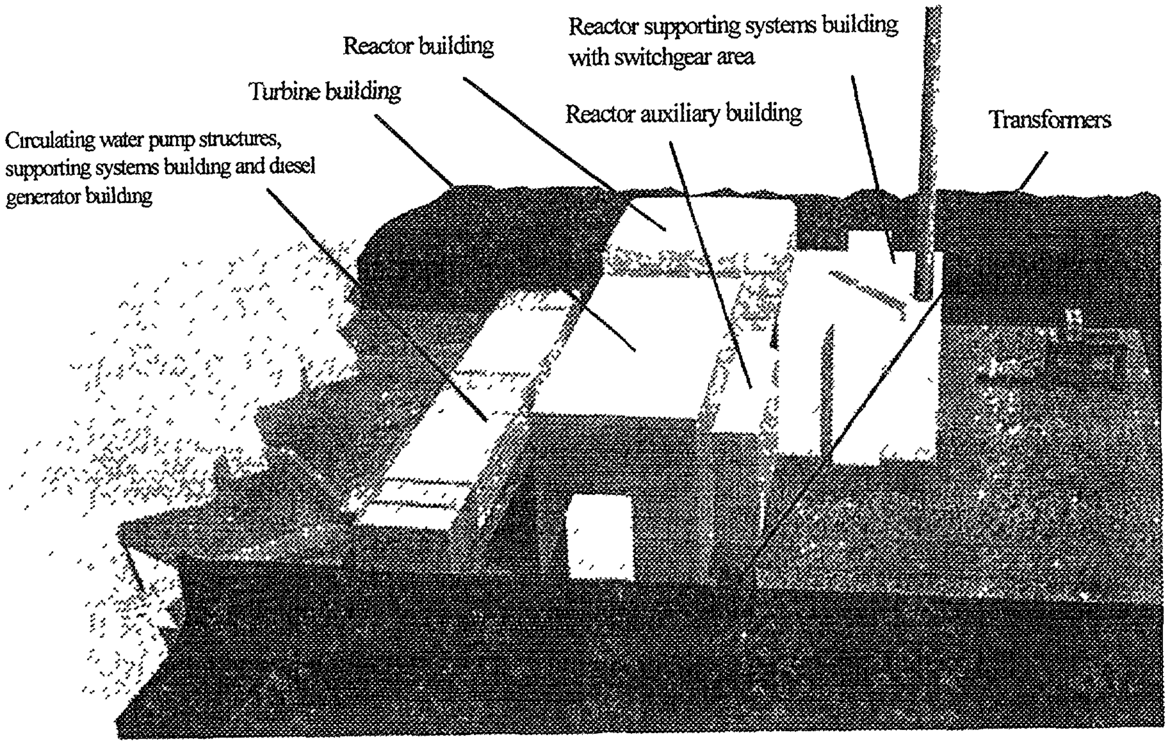


FIG. 17: Plant Site Layout

Operational I&C encompasses all systems required for process control in normal operation (i.e. during power operation and in the shutdown condition), such as:

- The process control system, including the operator station and emergency control station
- Automatic controls including protective devices and sensors in the plant
- Bus systems.

The task of the safety I&C system is to process and monitor key process variables important to reactor safety and environmental protection in order to detect accident conditions and to automatically initiate safety-related countermeasures in supplement to the passive switching actions in order to maintain reactor condition within safe limits. Safety I&C initiates no actions during normal undisturbed operation, but takes priority over all operational I&C system actions when required.

The computer-aided process information system provides a global information source. Intelligent information processing and compression enables it to display process conditions and process sequences with a high information content for safety-related and operational tasks.

The alarm and information system of the safety system first and foremost informs the operating staff of the status and condition of the safety system, and is used for service purposes.

The different I&C subsystems are connected via a common, redundant plant bus.

All accident conditions are detected by the safety I&C system, which initiates operation of the active safety equipment. If these safety systems fail, the passive safety equipment begins operating and brings the reactor to a safe condition.

As the passive safety features alone are capable of controlling all postulated accidents during power operation, the safety I&C system can also be limited to a redundancy of two 100%-capacity trains. Two 100 %-capacity I&C trains are therefore configured, the train allocation of which is maintained by I&C, process engineering and power supply systems. The process variables to be monitored are recorded by means of four measuring transducers. Further processing of measuring signals, limit value formation, formation and selection of actuation signals, and logic gating to form trip signals are all performed by means of a digital system. The trip signals directly actuate a process component or component group.

7 CIVIL STRUCTURES

7.1 General Arrangement of Buildings

The buildings are arranged in three complexes, thereby allowing simultaneous construction of all structures (Figure 17).

In terms of building arrangement, a distinction is made between site-dependent buildings such as the circulating/cooling water supply systems and ancillary systems building, and site-independent unit buildings, such as the following:

- Reactor building, including containment
- Turbine building
- Reactor auxiliary building
- Reactor supporting systems building, with switchgear area
- Diesel generator building
- Off-gas vent stack

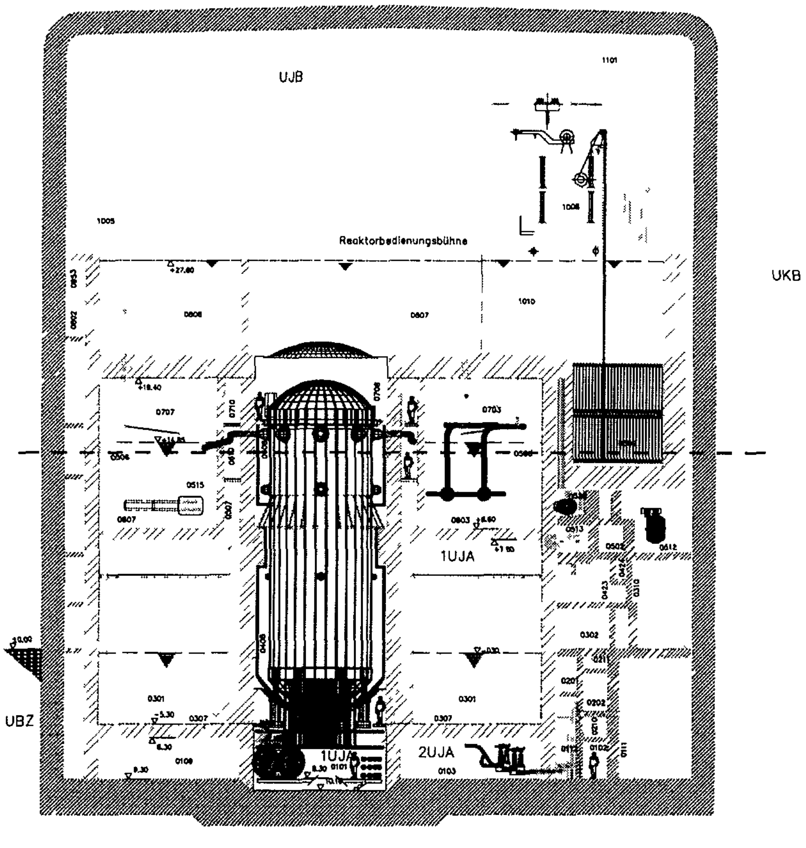


FIG. 18A: Reactor Building – vertical section

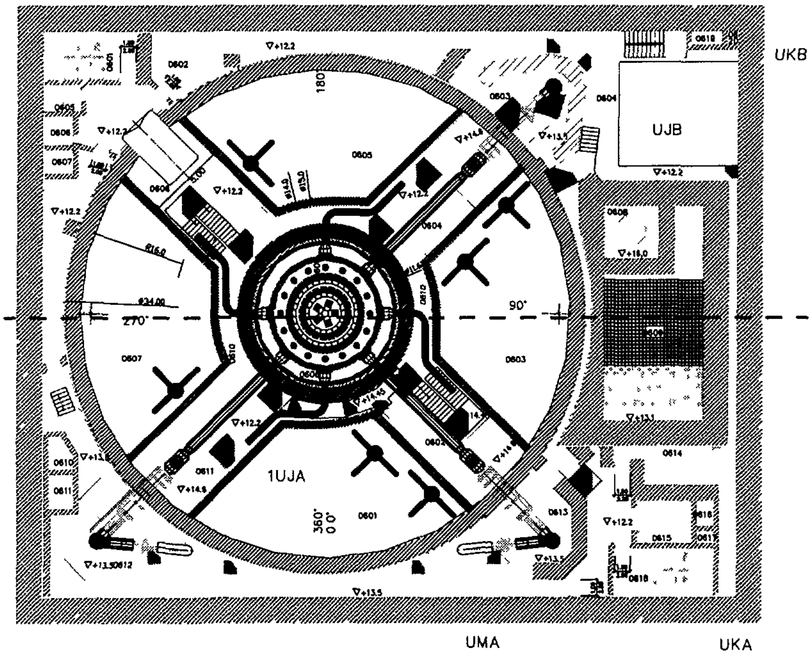


FIG. 18B: Reactor Building – horizontal section

The central unit comprises the reactor building and the turbine building, with partially integrated reactor auxiliary building. The reactor supporting systems building with the switchgear area and access to the controlled area are together connected to the central unit. Personnel entry and exit are exclusively via a stairwell, equipped with an elevator, which is centrally located in this building.

The circulating water pump house, the diesel generator building and the main conventional supporting systems are located at a distance from the unit buildings.

Piping and cables are routed either underground or in ducting structures.

7.2 Reactor Building

The reactor building (Figure 18A and B) houses the containment and the safety-related mechanical components, some of the electrical and I&C equipment, and the required power supply and protection systems. It provides protection against natural and external man-made hazards and ensures activity retention in the event of accidents.

The main systems and components located in the reactor building are the containment and its internals, the nuclear fuel storage and handling systems, the dryer-separator storage pool, the main steam and feedwater lines, the reactor water cleanup system, scram tanks, concentrate storage system and the heating, ventilation and air-conditioning systems, as well as the secured switchgear and the master control console.

The emergency control console is housed in the diesel generator building.

The structural concept is divided into three parts, as follows:

- Outer shell with penetration protection
- Inner structure, which is largely decoupled from the outer shell and the containment, and the
- Containment, whose structures, except for the base plate, are decoupled from the reactor building which encloses them.

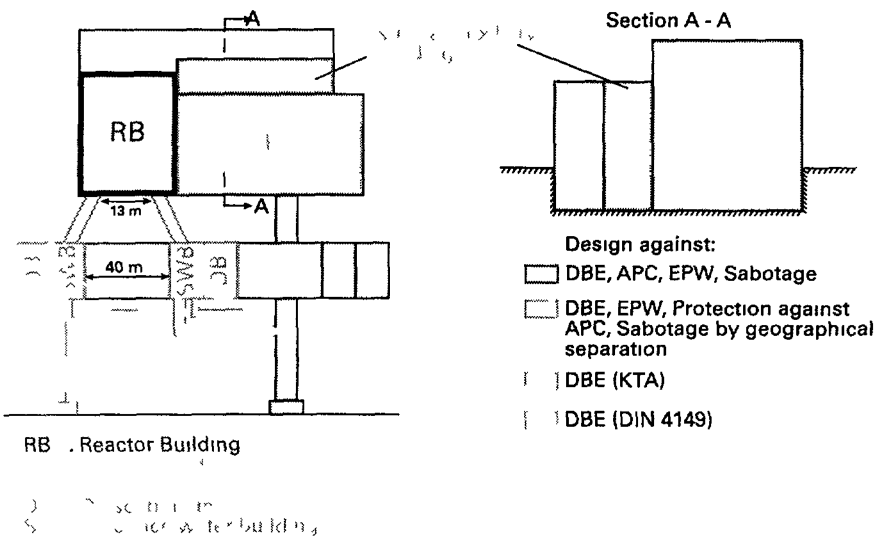


FIG 19: Plant Protection Against Natural and External Man-Made Hazards

7.3 Other Buildings

7.3.1 Turbine Building

The turbine building contains mainly the systems and components of the steam, condensate and feedwater cycle, with condensate-feedwater pumps and feedwater heater as well as the turbine and generator. The feedwater heating system consists of a single train, and the feed heaters are arranged vertically in the immediate vicinity of the turbine.

The turbine building forms part of the controlled area of the plant.

7 3 2 Reactor Auxiliary Building

The reactor auxiliary building contains systems and components for the treatment and storage of radioactive wastewater, including the evaporator system, as well as the filters of the main condensate cleanup system and the unsecured part of the closed cooling water system.

The selected arrangement of the main condensate cleanup system and wastewater treatment and storage system ensures short piping connections with the system and equipment areas in the turbine building.

7.3.3 Reactor Supporting Systems Building

The reactor ancillary systems building contains the workshops and parts of the waste treatment and storage system, as well as the central access point to the controlled area. This building houses components of the following systems:

- Intake and exhaust air system
- Sanitary facilities, in particular the changing rooms and washroom facilities required for access to the controlled area
- Laboratory
- Hot workshop and decontamination facilities
- Reserved space for mobile concentrate treatment with a drum store for low-active waste

7.3.4 Switchgear Building

As a result of changes in the electrical and I&C equipment concept, the switchgear in the SWR 1000 also differ from the previous standard arrangement in BWR plants. The switchgear building no longer exists as an autonomous structure. Instead, the switchgear are housed in two common stories of the reactor building and reactor supporting systems building. In addition to being centrally installed in this switchgear area, operational switchgear is installed on a localized basis where possible and appropriate.

7.4 Plant Protection Against Natural and External Man-Made Hazards

The plant is designed to withstand all natural and external man-made hazards as required under German licensing procedures, including, for example, the design basis earthquake (DBE), aircraft crash (APC), explosion pressure wave (EPW) and industrial sabotage (see Figure 19). The reactor building is thereby designed as part of a comprehensive protection concept to accommodate the postulated loads, and the systems and components located inside it are designed, as a function of their classification, to withstand the shocks induced. To reduce consequential loads inside the building, the building floors are horizontally decoupled from the external walls.

Both the diesel generator building and the circulating water pump house for the residual heat removal systems are designed to withstand a design basis earthquake and the loads due to an explosion pressure wave. The principle of physical separation (> 40 m) is applied to protect against aircraft crash. The cooling water and cable ducts are routed underground at an appropriate depth and at the required intervals (> 13 m).

LIST OF ABBREVIATIONS

A summary list of abbreviations used in the text comprises:

ATWS	anticipated transients without scram
BWR	boiling water reactor
CCC	containment cooling condenser
CRD	control rod drive
EC	emergency condenser
EdF	Electricité de France
EPR	European Pressurized Water Reactor
FW	Feedwater
GPR	Groupe Permanent Chargé des Réacteurs Nucléaires
GRS	Gesellschaft für Reaktorsicherheit
HP	high-pressure
I&C	instrumentation and control
LOCA	loss-of-coolant-accident
LP	low-pressure
LPCI	low-pressure core injection
MSIV	main steam isolation valve
MSL	main steam line
NPI	Nuclear Power International
PPPT	passive pressure pulse transmitter
PSA	probabilistic safety analysis
PWR	pressurized water reactor
RHR	residual heat removal
RPV	reactor pressure vessel
RSK	(German) Reactor Safety Commission
UKB	Reactor supporting systems building
UBZ	Ducting structure (cable)
1UJA	Containment

IAEA-SM-353/51

THE CANDU® 6

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Abstract

The CANDU® 6 is a modern nuclear power plant designed and developed under the aegis of Atomic Energy of Canada, Limited (AECL) for domestic use and for export to other countries. This design has successfully met criteria for operation and redundant safety features required by Canada and by the International Atomic Energy Agency (IAEA) and has an estimable record of performance in all applications to date. Key to this success is a defined program of design enhancement in which changes are made while retaining fundamental features proven by operating experience. Basic design features and progress toward improvements are presented here.

1. INTRODUCTION

The enhanced CANDU® 6 is a proven, up-to-date nuclear power plant design. It offers the attributes of a next-generation product, with a successful operating record implemented in recent build projects. AECL applies continuous improvement to the product evolution; incremental design improvements are incorporated in successive projects, while retaining fundamental features proven by operating experience. Thus the design and licensing bases are also updated, maximizing the benefits of real-world project and licensing experience.

Since 1996, AECL has conducted a systematic CANDU® 6 enhancement program. This has culminated in a design incorporating an integrated group of changes, each of which is practical as a stand-alone upgrade. This next-generation-enhanced design shares many common features with the family of plants; the following describes the main features and a number of the key improvements.

The CANDU® 6 is a nuclear power plant of the pressure tube type, notable for its use of heavy water as a coolant and moderator. Some of the design features and characteristics of the reactor are:

- a reactor core comprising 380 small diameter pressure tube fuel channels,
- heavy water (D_2O) for moderator and coolant,
- separate low pressure moderator and high pressure fuel cooling systems,
- on-power refueling,
- reactivity devices in the cool low pressure moderator, not subject to high temperatures or pressures,
- natural uranium fuel,
- reduced consequences from accidental reactivity fluctuations - excess reactivity available from the fuel is small, and the relatively long lifetime of prompt neutrons in the reactor precludes rapid changes in power levels, and
- special safety systems to shut down the reactor and isolate/retain radioactive fission products.

The design objectives are to meet current and evolving licensing requirements in Canada, to meet current and evolving IAEA safety standards and guides, and to include a comprehensive design definition adaptable to the host country's licensing requirements without fundamental changes.

1.1 Licensing basis

This plant meets the requirements of the Canadian regulatory agency, the Atomic Energy Control Board (AECB) and complies with International Atomic Energy Agency (IAEA) design guides. CANDU® 6 meets Canadian licensing regulations. There are two operating plants in Canada

(Point Lepreau and Gentilly-2) and units have been successfully licensed for operation in Argentina, Romania and South Korea including most recently, Wolsong-3.

1.2 CANDU® 6 operating characteristics

CANDU® stations operate extensively in the automatic, reactor following turbine mode, subjecting the plant to continuous small perturbations in reactor power with no adverse effects. Operating experience indicates that they can also operate in the load follow mode from 60 to 100 per cent of full power without adverse effects. The following summarizes operating characteristics:

The unit is capable of sustained operation at any electrical output up to 100 per cent of rated full power output. The normal operating mode is with reactor following turbine. For reactor power increases, the nuclear steam plant is capable of maneuvering at the following rates:

Power range	Rate
0 - 25 % of full power	4% of actual power per second
25 – 80 % of full power	1% of full power per second
80 –100 % of full power	0,15% of full power per second

1.3 CANDU® 6 performance

The first four generating units have over 50 years of cumulative operating experience and an excellent record. The plant at Point Lepreau, operated by New Brunswick Power, frequently led the world in lifetime capacity factor. The Wolsong-1, operated by the Korean Electric Power Corporation, was the best performing plant in the world for three out of its fourteen years of operation, including 1993, when it achieved a capacity factor of 100.8 per cent.

The average capacity factor in 1997 for all six plants operating in Canada, Korea and Argentina was 85.7 per cent. In fact, Wolsong-1, Wolsong-2 and Cernavoda-1 achieved capacity factors of 102, 97.1 and 89.1 per cent to the end of 1997, respectively. The average lifetime capacity factor for all operating CANDU® 6s was in excess of 85 per cent to the end of 1997.

2. DESCRIPTION OF THE NUCLEAR STEAM SUPPLY SYSTEM

2.1 Reactor assembly and fuel

2.1.1 Reactor Assembly

The CANDU® 6 reactor assembly comprises 380 channels contained in and supported by a horizontal cylindrical tank known as the calandria (see Figure 1). The calandria is closed and supported by end shields at each end. Each end shield comprises an inner and an outer tubesheet joined by lattice tubes at each fuel channel location and a peripheral shell. The inner spaces of the end shields are filled with steel balls and water, and are water-cooled. The fuel channels, supported by the end shields, are on a square lattice pitch. The calandria is filled with heavy water moderator at low temperature and pressure. The calandria is in a light water- filled, shield tank comprising a steel-lined, water-filled concrete vault. Horizontal and vertical reactivity measurement and control devices are set between and perpendicular to rows and columns of fuel channels. They operate in a low pressure, low temperature environment.

Each fuel channel contains 12 fuel bundles in the reactor core. The fuel channel assembly includes a zirconium 2.5 per cent niobium alloy pressure tube of internal diameter 103.38 mm, a zirconium calandria tube, stainless steel end fittings at each end, and spacers which maintain separation of the pressure tube and the calandria tube. The lattice pitch between fuel channels is 28.575 cm. The space between the calandria tube and the pressure tube is filled with $C0_2$ gas to insulate the hot pressure tube from the cool moderator. Each end fitting incorporates a feeder connection through which heavy water coolant enters and leaves the fuel channel. Pressurized heavy water coolant flows around and through the fuel bundles in the fuel channel and removes the heat generated in the fuel by nuclear fission.

During on-power refuelling, the fuelling machines gain access to the fuel channel by removing the closure plug and shield plug from both end fittings of the channel to be refueled.

2.1.2 Fuel

The standard CANDU® fuel bundle consists of 37 identical base elements, arranged in circular rings as shown in Figure 2. Each element consists of natural uranium in cylindrical pellets of sintered uranium dioxide contained in a zircaloy-4 sheath closed at each end by an end cap. The 37 elements are held together by end plates at each end to form the fuel bundle. Spacers brazed to the elements maintain the required separation of the fuel elements. The outer fuel elements have bearing pads to support the fuel bundle in the pressure tube.

Fuel performance has been excellent. While more than 900,000 bundles have been irradiated in Canada to date less than 0.1 per cent have developed defects. Similar performance has been achieved

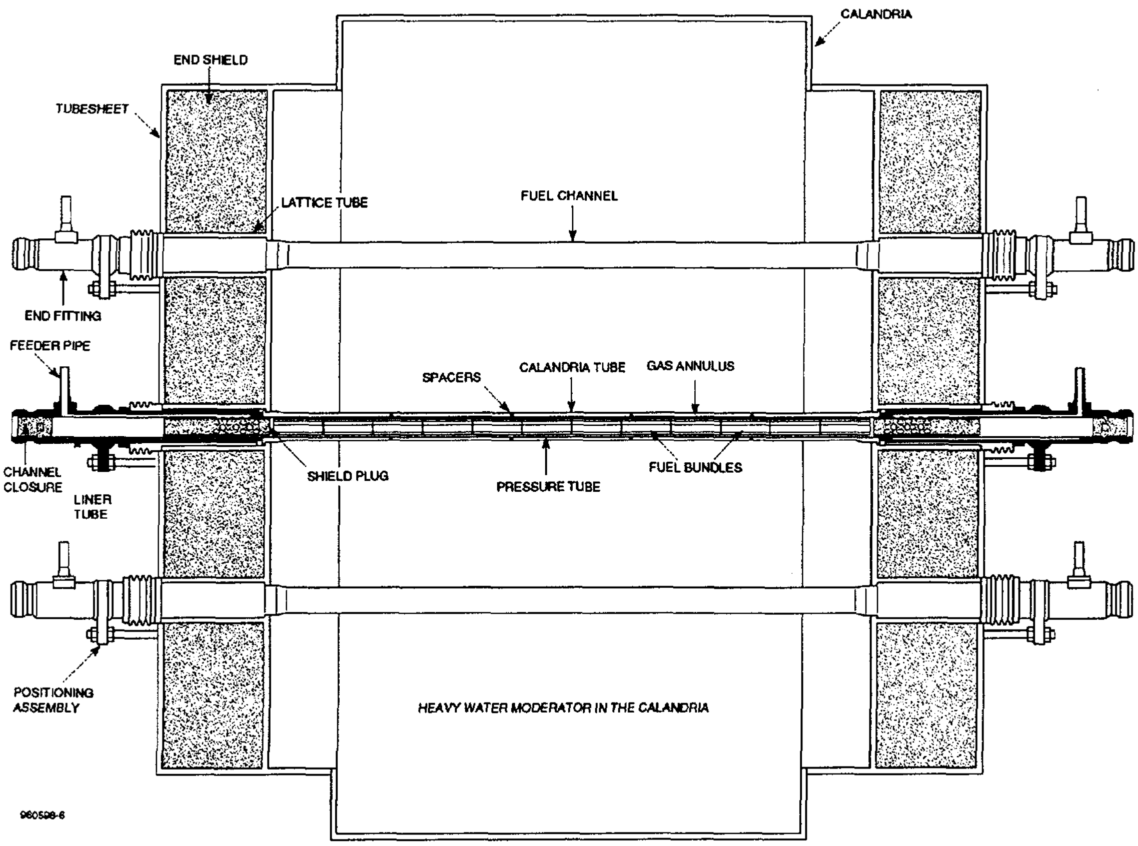

FIG. 1. Calandria Assembly Schematic.

in offshore units. Future design optimization will take advantage of the new CANFLEX 43-element fuel bundle. This fuel bundle uses graded pin sizing to achieve 20 per cent lower peak element rating, and 6-8 per cent greater thermal margins. The demonstration irradiation of CANFLEX fuel bundles is successfully under way at Point Lepreau station, New Brunswick, Canada.

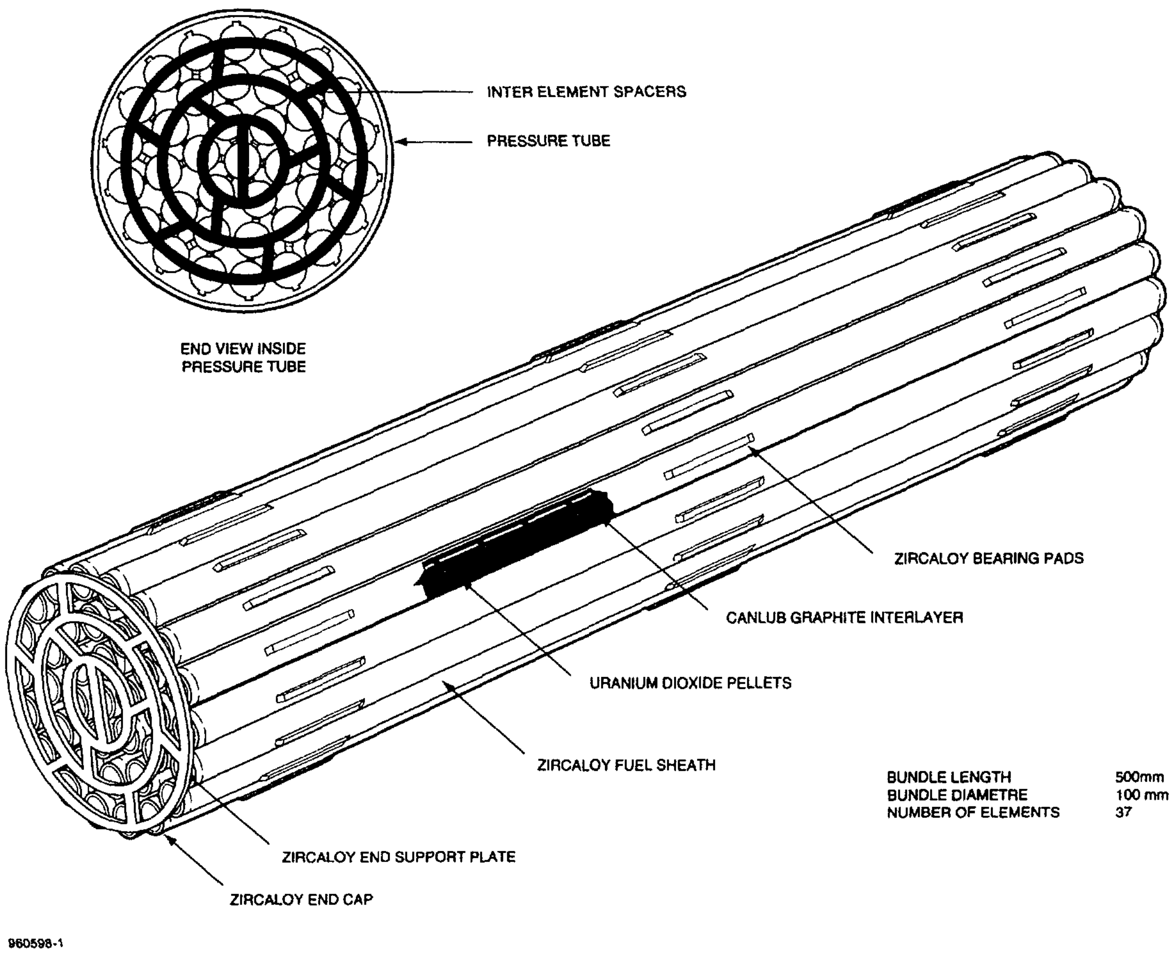

FIG. 2. 37-Element Fuel Bundle

2.2 Fuel Changing

The fuel changing operation combines use of two remotely controlled fuelling machines, one operating on each end of a fuel channel. New fuel bundles from one fuelling machine are inserted into a fuel channel in the same direction as the coolant flow and the displaced irradiated fuel bundles are received into the second fuelling machine at the other end. Either machine can load or receive fuel. The direction of loading depends upon the direction of coolant flow in the fuel channel being fuelled, which alternates from channel to channel. The fuelling machines receive new fuel while connected to the new fuel port and discharge irradiated fuel while connected to the discharge port. The entire operation is accomplished by a pre-programmed computerized system. The control system provides a printed log of all operations and permits manual intervention by the operator. Either all or part of the fuel bundles in a fuel channel can be changed at any visit.

2.3 Fuel handling system

The fuel handling system, illustrated in Figure 3, performs the following functions:

- provides facilities for the storage and handling of new fuel,
- refuels the reactor remotely while it is operating at any level of power, and
- transfers the irradiated fuel remotely from the reactor to the storage bay.

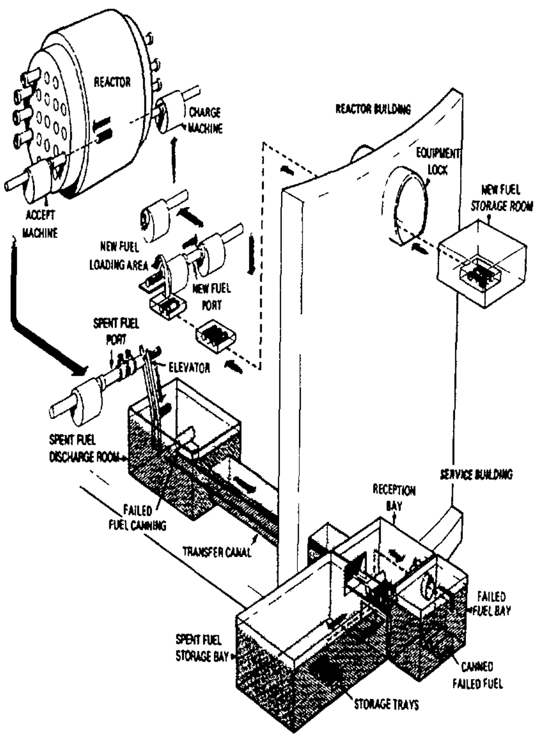

FIG. 3 Fuel Handling System Schematic.

2.4 Fuel Transfer

New fuel is received in the new fuel storage room in the service building. This room accommodates six months' fuel inventory and can temporarily store all the fuel required for the initial fuel loading. The fuel bundles are identified and loaded into the magazines of the new fuel ports. Transfer of the new fuel bundles into the fuelling machines is remotely controlled. Irradiated fuel received in the discharge port from the fuelling machine is transferred into an elevator, which lowers it into a water-filled discharge bay. The irradiated fuel is then conveyed under water through a transfer canal into a reception bay, where it is loaded onto storage trays or baskets and passed into the storage bay. The discharge and transfer operations are remotely controlled. Operations in the storage bays are carried out under water using special tools, aided by cranes and hoists. Defective fuel is inserted into cans under water to limit the spread of contamination. The storage volume of the bays has sufficient capacity for 20 years' accumulation of irradiated fuel.

2.5 Moderator and auxiliary systems

2.4.1 Moderator System

About four per cent of reactor thermal power appears in the moderator. The largest portion of this heat is from gamma radiation; additional heat is generated by moderation (slowing down) of the fast neutrons produced by fission in the fuel, and a small amount of heat is transferred to the moderator from the hot pressure tubes. The moderator system is illustrated in Figure 4. The system includes two 100 per cent capacity pumps, two 50 per cent flow capacity heat exchangers cooled by recirculated cooling water and a number of control and check valves. Connections are provided for the purification, liquid poison addition, heavy water (D_2O) collection, supply and sampling systems. An enhancement to the CANDU® 6 design includes recycle of the moderator waste heat through low-pressure feedwater heating.

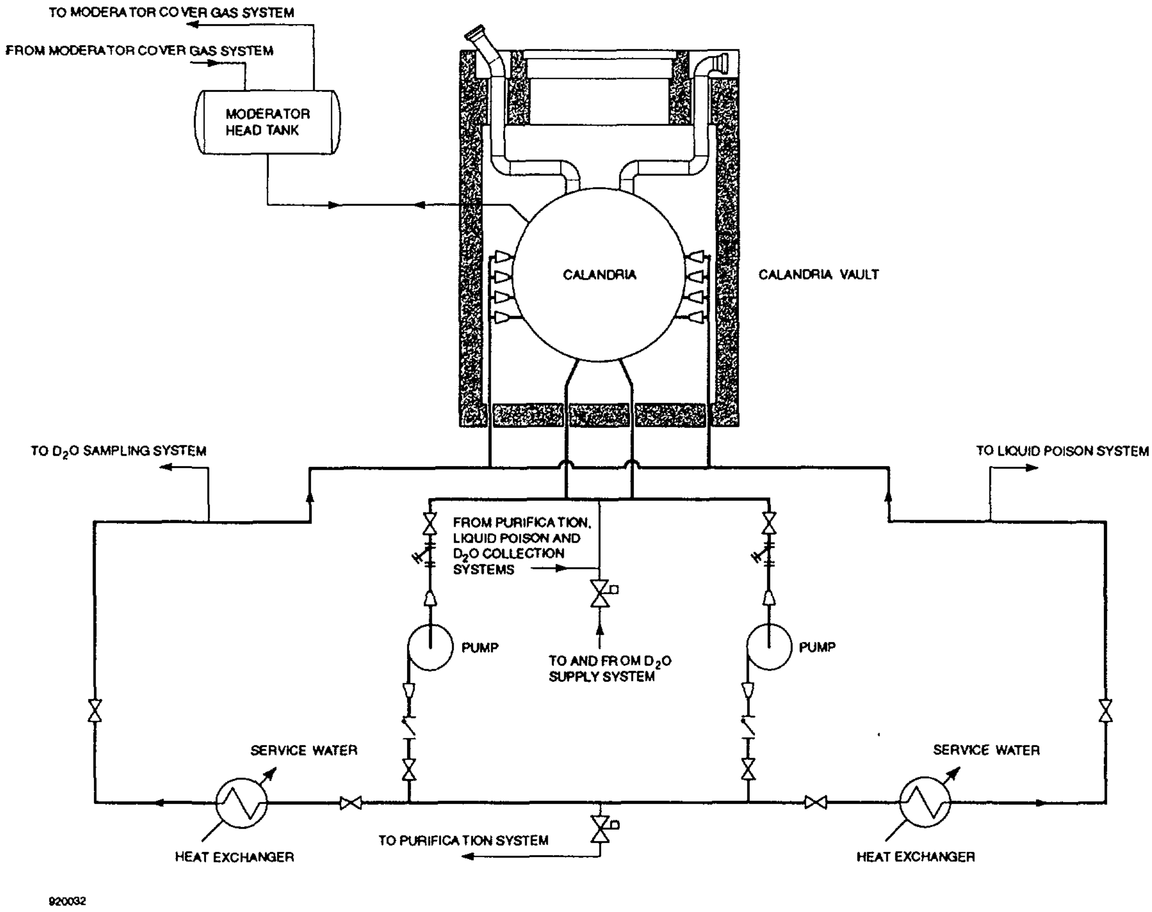

FIG. 4. Moderator System

The moderator pump motors are connected to the medium voltage Class III power supply. In addition, each pump has a pony motor, capable of driving the pump at 25 per cent speed, connected to the Class II power supply. In the event of a loss of Class IV power, standby diesel generators are available to supply Class III power.

The heavy water in the calandria functions as a heat sink in the unlikely event of a loss of coolant accident in the heat transport system coincident with a failure of emergency core cooling.

2.5.1 Cover Gas System

Helium is used as the cover gas for the moderator system because it is chemically inert and is not activated by neutron irradiation. Radiolysis of the heavy water moderator in the calandria results in production of deuterium and oxygen gases. The cover gas system prevents accumulation of these gases by catalytically recombining them to form heavy water. The moderator cover gas pressure is 27 kPa(g) in normal operation.

2.5.2 Purification System

The moderator purification system:

- maintains the purity of D_2O, thereby minimizing radiolysis which can cause excessive production of deuterium in the cover gas,
- minimizes corrosion of components, by removing impurities present in the D_2O and by controlling the pD (pH),

- under operator command, reduces the concentration of the soluble poisons, boron and gadolinium, in response to reactivity demands, and

- removes the soluble poison, gadolinium, after shutdown system number 2 has operated.

2.5.3 Operation of the Moderator System

The series/parallel arrangement of the system lines and valves permits the output from either pump to be cooled by both of the heat exchangers and assures an acceptable level of moderator cooling when either of the two pumps is isolated for maintenance. Reactor power is reduced to about 60 percent if one moderator heat exchanger is isolated.

2.6 Heat transport system and auxiliary systems

2.6.1 System Operation

The heat transport system (HTS) circulates pressurized D_2O coolant through the fuel channels and transports the heat to steam generators, where it is transferred to light water to produce steam to drive the turbine. Two parallel HTS coolant loops are provided in CANDU® 6. The heat from half of the 380 fuel channels is removed by each loop. Each loop has one inlet and one outlet header at each end of the reactor core. D_2O is fed to each of the fuel channels through individual feeder pipes from the inlet headers and is returned from each channel through individual feeder pipes to the outlet headers. Each heat transport system loop is arranged in a "Figure 8" configuration, with the coolant making two passes, in opposite directions, through the core during each complete circuit. The HTS piping is fabricated from corrosion resistant carbon steel. Figure 5 is a schematic of the heat transport system. The heat transport system operates at a coolant outlet header pressure and temperature of

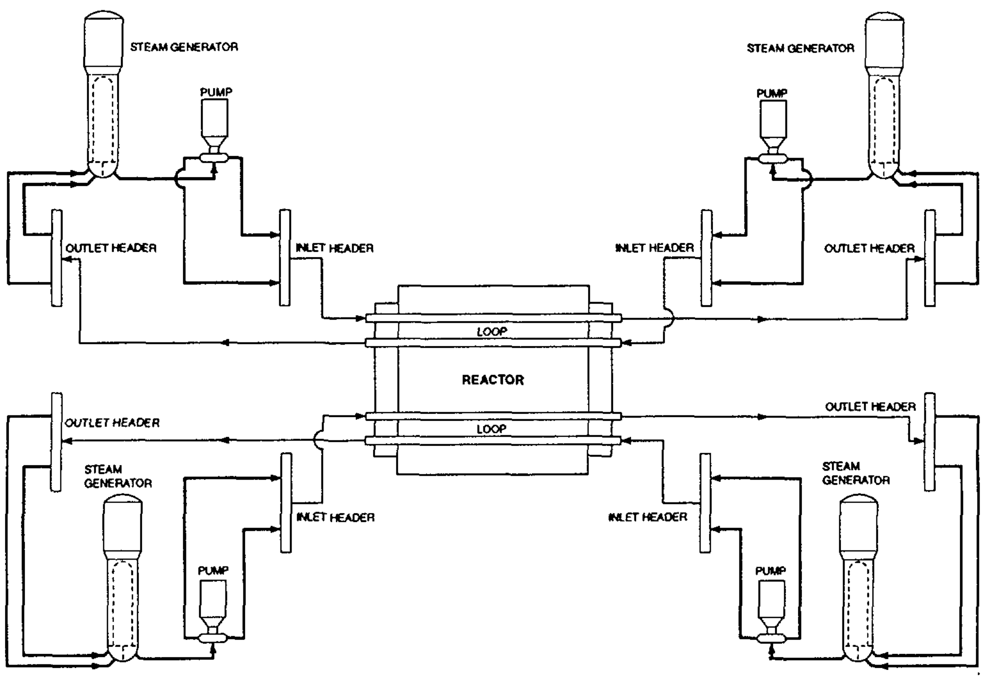

FIG. 5. Heat Transport System

10.0 MPa(a) and 310°C at design conditions. Total heat transport coolant flow rate is 7.7 Mg/s. Flows in individual fuel channels are optimized by choice of feeder pipe design to maintain similar coolant outlet conditions throughout the core. Heat transport system materials are chosen for protection against corrosion effects to ensure long design life and also capability for life extension up to 60 years.

The pressure in the heat transport system is controlled by a pressurizer connected to the outlet headers at one end of the reactor. Valves provide isolation between the two loops in the event of a loss-of-coolant accident. The principal performance features for the heat transport system and associated systems are as follows:

- reactor coolant circulates through the fuel channels at all times during reactor operation, shutdown and maintenance,
- HTS pressure is controlled in normal modes of operation by the pressure and inventory control system,
- the HTS is protected from overpressure by instrumented relief valves and by the reactor regulating and safety system,
- HTS coolant inventory is controlled in normal modes of operation by pressure and inventory control systems,
- the shutdown cooling system, capable of operation at full HTS temperature and pressure, is used to remove reactor decay heat during shutdowns and maintenance outages; this system also permits the draining of pumps and steam generators in the HTS for maintenance,
- purification by filtering, ion exchange and degassing, and chemical addition maintains the chemistry and purity of the reactor coolant, and
- by using welded construction and bellows-sealed valves where practical, heavy water leaks are minimized. Potential leak sources are connected to the heat transport D_2O collection system.

2.6.2 Heat transport auxiliary systems

2.6.2.1 Pressure and Inventory Control System

The heat transport pressure and inventory control system consists of a pressurizer, a degasser - condenser, D_20 feedpumps, feed and bleed valves and a D_2O storage tank (see Figure 6). The system provides pressure and inventory control and overpressure protection for the HTS. The principal system functions are to:

- control HTS pressure over the full range of HTS and reactor operating modes,

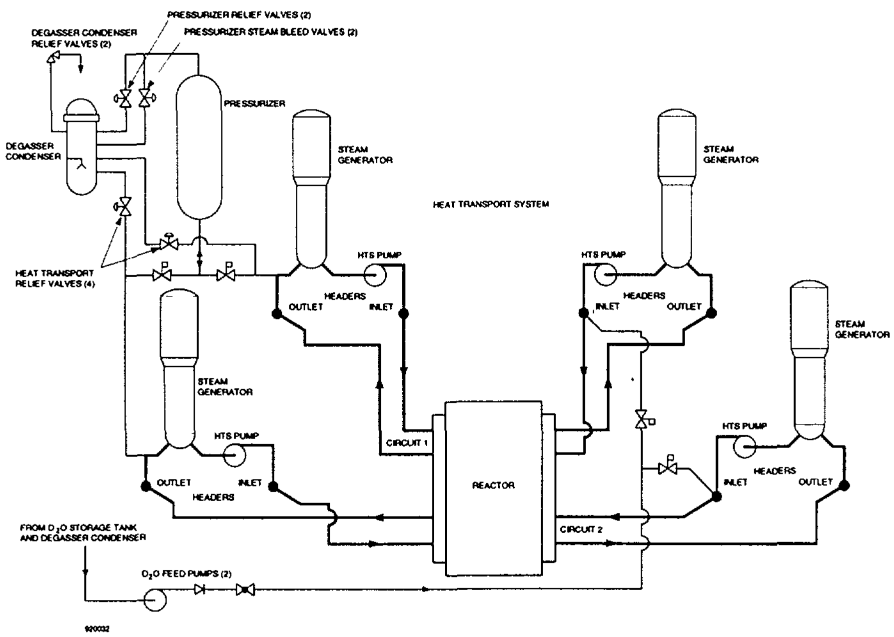

FIG. 6. Heat Transport Pressure and Inventory Control System

- control HTS inventory over the full range of HTS and reactor operating modes,
- limit HTS pressure increases or decreases caused by transients to acceptable values,
- accommodate HTS coolant thermal expansion and contraction associated with warm-up, cool-down, start-up, shutdown, and power maneuvering, and
- provide degassing of the HTS coolant.

2.6.2.2 Shutdown Cooling System

The shutdown cooling system removes decay heat following a reactor shutdown and cools the HTS to a temperature suitable for maintenance of heat transport and auxiliary systems components. It can cool the HTS from full system pressure and temperature and maintain it at a low temperature for an indefinite period time. It can also provide fuel cooling with the HTS coolant drained to header level, to facilitate maintenance and inspection of steam generators and/or heat transport pump internals.

The shutdown cooling system consists of pumps and heat exchangers arranged in two separate circuits. In each circuit, one at each end of the reactor, the pump takes coolant from the reactor outlet header and returns it to the reactor inlet header via the heat exchangers. The pumps and heat exchangers are below the reactor headers so a net positive suction head is available for the pumps when the HTS drains to the headers.

The shutdown cooling system is normally cold, depressurized, and isolated from the HTS by valves during reactor operation. The shutdown cooling pumps are provided with backup power from the Class III power supply. During normal operation, the shutdown coolers are cooled by the recirculated cooling water system.

The D_2O collection system collects any heavy water leakage from mechanical components, and receives heavy water drained from equipment prior to maintenance. Collected heavy water is returned to the heat transport heavy water storage tank.

2.6.2.3 Purification System

The heat transport purification system minimizes the accumulation of active deposits within the system. The production of radioactive materials in the main HTS is very low. This is due to restrictions on materials used in the system (for example, only very low cobalt contents are permitted in system materials). This is also due to the absence of failed fuel during reactor operation (if a fuel failure occurs, it is detected and promptly removed by the on-power refuelling system). To minimize the accumulation of active deposits within the HTS, the coolant is continuously filtered and purified. The pressure head of the HTS pump in each loop allows a flow of HTS coolant through the purification system. This flow is cooled prior to passage through filters and ion exchange columns and then returned to the main HTS via an interchanger, which tempers the purification flow and minimizes heat loss.

2.7 Reactor regulating system

2.7.1 Reactor regulating system reactivity control devices

The basic requirement of the reactor regulating system (RRS) is to control reactor power at a specified level and, when required, to maneuver the level between set limits at specific rates. The reactor regulating systems include:

- light water zone control absorbers,
- mechanical control absorbers,

- adjusters, and
- soluble poison to the moderator.

2.7.1.1 Light Water Zone Control Absorbers

Light water (H_2O) is a neutron absorber (poison) in the heavy water CANDU® reactor. The liquid zone control system takes advantage of this to provide short-term global and spatial reactivity control in the reactor core. The liquid zone control system in the reactor consists of six tubular, vertical, zone control units spanning the core. Each zone control unit contains either two or three zone control compartments, a total of 14 zone control compartments in the reactor. The zone control units are located such that the 14 zone control compartments are distributed throughout the core for flux control. The addition or removal of light water to/from the liquid zone control compartments controls flux (power) in each zone. The total reactivity worth of the zone control obsorbers is 7.5 mk under normal operation.

2.7.1.2 Mechanical Control Absorbers

Four mechanical control absorbers, mounted above the reactor, can be driven in or out of the core at variable speeds, or dropped by gravity into the core, between columns of fuel channels, by releasing a clutch. These absorbers are normally parked out of the core; they are driven in to supplement the negative reactivity from the light water zone control absorbers, or are dropped to effect a fast reduction in reactor power. When inserted, the mechanical control absorbers also help to prevent the reactor from going critical when the shutoff rods of shutdown system 1 are withdrawn, and are interlocked, in this inserted position until shutdown system number 1 is energized and available.

2.7.1.3 Adjusters

Adjusters are cylindrical neutron absorbing rods. A CANDU® reactor typically has 21 vertically mounted adjuster rods, normally stainless steel, normally inserted between columns of fuel channels for flux shaping purposes. Removal of adjusters from the core provides positive reactivity to compensate for xenon buildup following large power reductions or in the event that the on-power refueling system is unavailable. The adjusters are driven in and out of the reactor core at variable speeds to provide reactivity control. The adjusters are normally driven in banks, the largest bank containing five rods. Total adjuster reactivity worth is 15 mk. A design enhancement to optimize adjuster positioning in the core for increased burn-up has been defined.

2.7.1.4 Poison Addition and Removal

A reactivity balance can be maintained by addition of soluble poison to the moderator. Boron is used to compensate for excess reactivity when fresh fuel is introduced. Gadolinium is added when the xenon load is significantly less than equilibrium (as happens after prolonged shutdowns). An ion exchange system removes the poisons from the moderator. The reactor regulating systems are controlled by a set of computer programs, which process the inputs from the reactor instrumentation and activate the appropriate devices.

2.7.2 Nuclear instrumentation

The nuclear instrumentation systems are designed to measure reactor neutron flux over the full operating range. These measurements are inputs to the reactor regulating system and safety systems. Instrumentation for the safety systems is independent of that utilized by the reactor regulating system. The principal instrumentation utilized for reactor regulation includes:

- an ion chamber system, and
- a self-powered, in-core, flux detector system.

2.7.2.1 Ion Chamber System

Three ion chambers are employed in the reactor regulating system, for measuring neutron flux in the range from 10^{-7} to 15 per cent of full power. These ion chambers are in housings at one side of the core. In addition to one ion chamber for the reactor regulating system, each housing also contains an ion chamber and shutter for shutdown system number 1. Each of the three channels consists of an ion chamber and amplifier unit. The solid state amplifiers upgrade the ion chamber outputs to suitable input signal levels for processing in the control computers. Three similar ion chambers, mounted on the other side of the core, provide inputs to shutdown system number 2.

2.7.2.2 Self-powered In-core Flux Detectors

In the high power range (above 15 per cent power), self powered in-core flux detectors provide detailed spatial flux information throughout the reactor core. Two types of in-core detectors are used; one type uses platinum as the sensitive emitter material, while the other uses vanadium. The platinum detectors are fast acting, sensitive to both neutrons and gamma rays, and because of their prompt response to flux changes, are used in the reactor regulating system and in the two shutdown systems. The vanadium detectors are sensitive to neutrons, but because of a relatively slow response to flux changes, are used only in the flux mapping system.

The in-core flux detectors of the regulating system and of shutdown system number 1 are mounted vertically in the core, while those of shutdown system number 2 are mounted horizontally. These in-core flux detectors provide exceptional capability to detect not only bulk power deviation but small-scale local power changes. Incorporating operating experience into detector position optimization is part of the CANDU® 6 enhancements program.

2.7.2.3 Start-Up Instrumentation

Special temporary instrumentation is required during the initial reactor start-up to monitor the neutron flux over the range from the spontaneous fission flux level to the sensitivity level of the permanent ion chambers. After start-up, this instrumentation is removed and is not required for subsequent start-ups. Two sets of triplicate fission chambers are used; one set covers the very low flux range (10^{-14} to 10^{-10} of full power). The second set covers the flux range from 10^{-11} to 10^{-6} of full power and overlaps the ranges of both the first set and the permanent ion chamber instrumentation.

2.7.2.4 Reliability

The reliability of the reactor regulating system is of paramount importance and is achieved by:

- direct digital control from dual redundant control computers,
- self-checking and automatic transfer to the standby computer on fault detection,
- control programs independent of each other,
- duplicated control programs,
- duplicate and triplicate inputs, and
- hardware interlocks limiting the amount and rate of change of positive reactivity devices.

Recently-demonstrated improvements to signal noise analysis provide further aid to operator monitoring of nuclear instrumentation performance.

2.7.2.5 Safety hardware interlocks

The reactivity mechanisms are subject to a number of interlocks external to the control computers, limiting the consequences of erroneous operation. When the reactor is in a tripped state (i.e., shutdown system 1 and/or shutdown system 2 inserted) interlocks prevent withdrawal of the adjusters and mechanical control absorbers. Poison removal from the moderator is also inhibited to prevent increased reactivity. The interlocks remain active, preventing reactor startup, until shutoff rods are fully withdrawn and available for reactor shutdown. There are further interlocks to prevent more than a limited number of high worth adjusters from withdrawal at the same time. This limits the rate of positive reactivity insertion.

2.8 Feedwater and main steam system

2.8.1 Feedwater System

The main feedwater system generally includes three 50 per cent capacity electrically driven main feedwater pumps that take suction from the deaerator storage tank and additional auxiliary feedwater supply. Control valves regulate flow to each steam generator. An auxiliary feedwater pump is provided to supply four per cent of full power feedwater requirements during shutdown conditions, or if the main feedwater pumps become unavailable.

Demineralization, de-aeration, oxygen scavenging and pH control feedwater chemistry. A blow-down system is provided for each steam generator to remove impurities that collect in the steam generators to prevent possible long-term corrosive effects.

2.8.2 Steam Generators and Main Steam Systems

The steam generators are of the shell and tube type design (Figure 7). Heavy water coolant flows through the small inverted U-tube bundles, transferring heat to the ordinary water to produce steam. Moisture is removed from the steam by cyclone separators in the upper section of the steam generator. The steam then flows via four separate steam mains, through the reactor building wall to the turbine where they connect to the turbine steam chest.

The turbine governor valves that admit steam to the high-pressure stage of the turbine, control steam pressure. If the turbine is unavailable, up to 70 per cent of full power steam flow can bypass

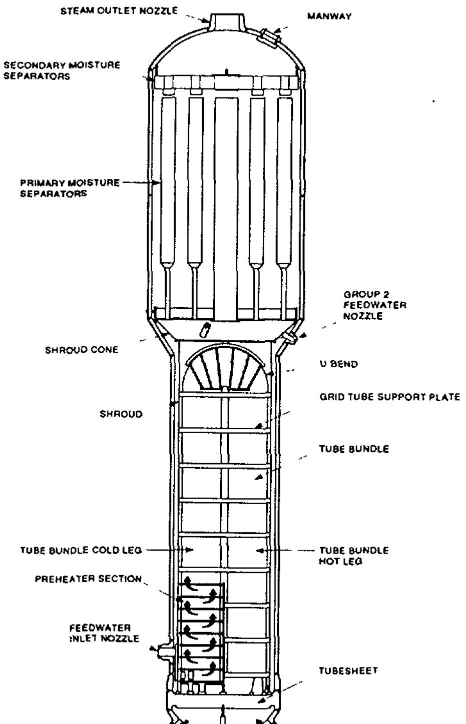

FIG. 7. Steam Generator

the turbine and go directly to the condenser. During this operation, the turbine by-pass valves control pressure. Auxiliary bypass valves are also provided to permit up to 10 per cent of full power steam flow during low power operation. Improved steam generator design with a larger steam drum has been introduced into CANDU 6 to improve response to transients and to increase time for operator response.

Steam pressure can also be controlled by discharging steam directly to the atmosphere via four atmospheric steam discharge valves (ASDV) which have a combined capacity of 10 per cent of full power steam flow. These valves are used primarily during warm-up or cool-down of the heat transport system. Four safety relief valves connected to the four main steam lines provide over-pressure protection for the steam system.

3. TURBINE GENERATOR SYSTEM

The turbine generator system consists of a turbine, generator, moisture separator-re-heaters, exciter, controls, and auxiliary subsystems. Steam flow to the turbine, and therefore the governor valves whose opening depends on turbine load setpoint and turbine speed, determine turbine power. The governor droop, i.e. frequency error to stroke governor valves fully, is typically 4 per cent but may be increased during synchronization. The turbine load setpoint can be raised or lowered at several rates. The slower rates are used for normal load maneuvers; the fast rates unload the set quickly during upset conditions such as reactor trips.

3.1 Steam Turbine

The steam turbine is a tandem compound unit, directly coupled to an electrical generator by a single shaft. It comprises one double flow; high-pressure cylinder followed by external moisture separators, live steam re-heaters and either two or three double flow, low pressure cylinders. The turbine system has main steam stop valves, governor valves; reheat intercept and emergency stop valves. All of these valves close automatically in the event of a turbine protection system trip.

3.2 Generator

The electrical generator is a three-phase, four-pole, gas cooled machine running at synchronous speed. The excitation unit consists of a solid-state, automatic voltage regulator controlling a thyristor converter that supplies the generator field via a field circuit breaker, generator slip rings and brush gear. The main power output from the generator to the main output step-up transformer is by means of a forced air cooled, isolated phase bus duct, with tap-offs to the unit service transformer, excitation transformer and potential transformer cubicle.

3.3 Condensing System

The condenser is the heat sink for the steam cycle. During normal operation it receives and condenses the exhaust from the turbine generator. During abnormal operation it receives the bypass steam from the condenser steam discharge valves (turbine bypass valves).

The turbine condenser is designed with three separate shells. Each shell is connected to one of the three low-pressure turbine exhausts. Exhaust steam from the turbine entering the shell is condensed by flowing over a tube bundle assembly of cooling water, typically water from a lake, river or ocean. A vacuum system removes air and other non-condensable gases from the condenser shells. The condenser is designed to accept turbine bypass steam to permit the reactor power to be reduced from 100 per cent power to 70 per cent if the turbine is unavailable. The bypass can accept 100 per cent steam flow for a few minutes, and 70 per cent of full power steam flow continuously.

3.4 Feedwater Heating System

The regenerative feedwater heating system heats the condensate to the required temperature before supplying it to the steam generators. The system comprises the feedwater heating equipment, the extraction steam system, and the feedwater heater drains and vents systems.

4. ELECTRIC POWER SYSTEM

4.1 Output System

The electric power system comprises a main output transformer, unit and system service transformers, and a switchyard. The output transformer steps up the generator output voltage to match the grid requirements for transmission to the load centers and also supplies the power to operate all of the station services. The switchyard contains equipment to permit switching of outputs between transmission lines, comprising circuit breakers, switches, lightning and grounding protection equipment to shield against electrical surges and faults.

4.1.1 Main Transformer

The main transformer steps up the generator output voltage to the transmission voltage. The transformer is rated to meet the generator output requirements and electrical grid parameters. It is equipped with standard accessories and protective devices.

4.1.2 Switchyard

The switchyard is the interface between the station and the power grid transmission lines. There are at least two incoming lines that are synchronized under normal conditions. However, the switchyard electrical equipment allows transmission of full station power through any one of the incoming lines.

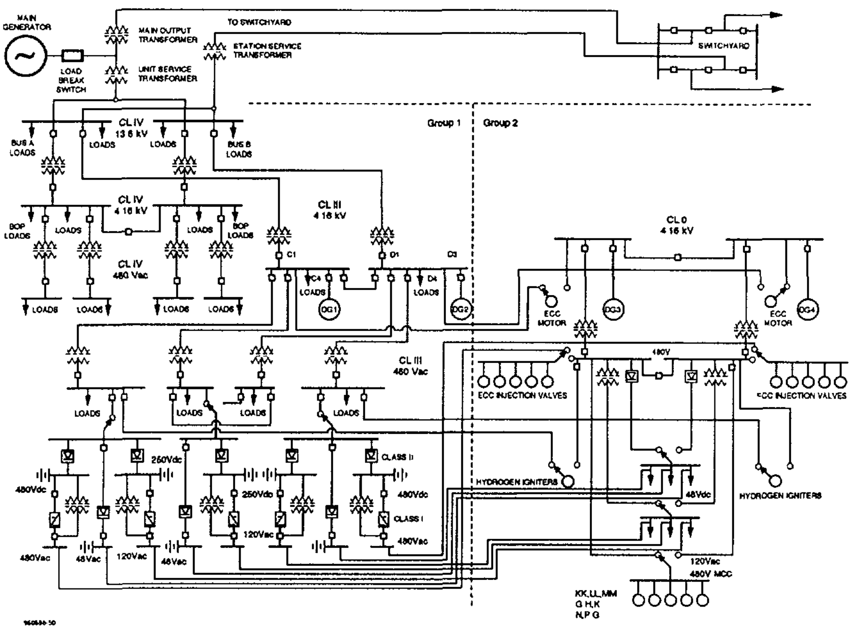

FIG. 8. CANDU 6 Single Line Diagram

4.1.3 Unit Service Transformer

Both the unit service transformer and the system service transformer supply the station services power during normal station operation. However, either transformer can provide the total service load in the event one is unavailable. The transformer is fed from the output system of the turbine generator.

4.1.4 System Service Transformer

The system service transformer is similar to the unit service transformer. It supplies half of the plant services power requirements under normal operating conditions and can provide the total service load when necessary. This transformer is fed from the grid and supplies all plant loads during the start-up of the plant, or when the turbine generator is unavailable. Both the unit service and station service transformers are designed to automatically maintain voltage in the station services.

4.2 Electric Power System Station Services

4.2.1 Power Classification

The station services power supplies are classified in order of the levels of reliability required. The reliability requirement of these power supplies is divided into four classes ranging from uninterruptible power to that which can be interrupted with limited and acceptable consequences. The CANDU® 6 station services single line diagram is shown in Figure 8.

4.2.1.1 Class IV Power Supply

Class IV power is alternating current, available from the grid or turbine-generator, serving as the primary power source to the station. Long term interruption of Class IV power can be tolerated without endangering equipment, personnel, or safety. Some Class IV loads include the main heat transport pumps, feed pumps, circulating water pumps and extractor pumps. Complete loss of Class IV power will initiate a reactor shutdown.

4.2.1.2 Class III Power Supply

Class III alternating current is fed from either Class IV power or on-site standby diesel generators. It supplies loads necessary for the safe shutdown of the reactor and turbine. The maximum interruption time of Class III power is 180 seconds. Some Class III loads include pumps in the service water system, emergency core cooling system, moderator circulation system, shutdown cooling system, heat transport system feed lines, steam generator auxiliary feed line, and the air compressors and chillers.

4.2.1.3 Class II Power Supply

Class II power is uninterruptible alternating current power supplies derived from inverters connected to Class 1 batteries in parallel with Class III rectifiers. It supplies all essential AC loads that cannot tolerate loss of power. Some Class II loads include critical motors, safety systems equipment, emergency lighting, computers and instrumentation.

4.2.1.4 Class I Power Supply

Class I power is uninterruptible direct current power supplies derived from batteries connected in parallel with Class III rectifiers. Class I power supply loads include essential DC motors, safety systems equipment, logic circuits and switchgear control circuits.

4.2.2 Automatic Transfer System

To ensure continuity of supply, in the event of a failure of either the unit or system service transformers, an automatic transfer system is incorporated in the station service buses. Transfer of load from one service transformer to the other is accomplished by:

- a parallel transfer of power under normal operating conditions for maintenance purposes,
- a fast open transfer of power initiated by the electrical protection system; this ensures that the voltage vector between the incoming supply and the residual voltage on the motors is small, so that the inrush current and the transient torque are kept to a small value,

- a residual voltage transfer, comprising automatic closure of the alternate breaker after the residual voltage has decayed by approximately 70 per cent; this is time delayed, may require load shedding and could result in reactor power cut-back. It is provided as a backup to the above transfers.

4.2.3 Station Batteries

The station batteries are normally on continuous charge from the Class III power supply and sized to supply Class I loads during a power interruption for the mission time.

4.2.4 Standby Generators

Standby power for Class III loads is supplied by two diesel generator sets housed in separate rooms with fire resistant walls. Each diesel generator can supply the total safe shutdown load of the unit. The Class III shutdown loads are duplicated, one complete system being fed from each diesel generator. In the event of failure of Class IV power, the two diesel generators start automatically. The generators can be brought up to speed and ready to accept load in less than two minutes. Each generator automatically energizes half of the shutdown load through a load-sequencing scheme. There is no automatic electrical tie between the two generators, nor is there a requirement for synchronization. In the event one generator fails to start, the total load can be supplied from the other generator.

4.2.5 Emergency Power Supply System

The emergency power supply system provides shutdown electrical loads essential for safety. The system and its buildings are seismically qualified to operate after an earthquake. The system is backup for one group of safety systems (shutdown system number 2, emergency water supply, secondary control area) if normal electric supplies become unavailable or the main control room becomes uninhabitable. The system comprises two diesel-generating sets in separate fire resistant rooms, completely independent of the station's normal services.

5. STATION INSTRUMENTATION AND CONTROL

5.1 Instrumentation and Control

The primary unit instrumentation and control systems are designed to give the operators in the main control room all information and control capabilities to operate the unit safely during normal and abnormal operation. A separate, independent control and instrumentation system, in a secondary control area, is designed for shutting down the unit and maintaining it in a shutdown condition in case the main control room becomes uninhabitable.

The main control room features extensive use of computer-driven color-graphic CRT (Cathode Ray Tube) displays, and offers selective presentation of information in diagrammatic formats. The use of CRT displays, designed using modern human factors engineering, simplifies the clutter of typical control room panels, and provides a uniform man-machine interface for all plant systems. Plant surveillance is also centralized in the main control room. For example, instrumentation for closed-circuit television, meteorological sensing, fire detection and alarm vibration monitoring and access control are all indicated and controlled from the main control room. A number of communications networks such as telephone, public address, maintenance, and plastic suit are also centralized here.

A major feature of all CANDU® generating stations is an integrated control philosophy using direct digital control. The plant is highly automated to require a minimum number of operator actions during all phases of operation. Control of main systems, such as the turbine, steam generator, reactor, heat transport and moderator systems are all under the control of the plant computer systems. This permits reliable and practical application of multivariable control algorithms for dynamic compensation in systems where this is advantageous.

The dynamic control of reactivity mechanisms by the plant computers is a prime feature of CANDU® reactors Through the manipulation of mechanical control absorbers, adjusters, and light water zone controllers, active spatial control of neutron flux within the reactor is continuously applied for optimum performance. Direct digital control of the reactor also incorporates two automatic power reduction algorithms upon detection of abnormal conditions. These algorithms force automatic, controlled, power reductions under certain conditions, to avoid tripping the unit by shutdown systems

5.2 Control Room

The latest CANDU® 6 design features the advanced control room, illustrated in Figure 9. It features an array of panels at the perimeter with two large central display screens, and the operations console Information on the panels and at the operations console allow safe control and monitoring of the station. The instrumentation and controls on the panels are grouped by system, with a separate panel allocated to each major system. Color monitors (CRTs) and advanced annunciation systems provide uncluttered control room panel layouts and excellent monitoring capabilities. The operator can call up information displays on the panel CRTs, the operating control console CRTs, and central display screens in a variety of alphanumeric and graphic formats via keyboards All display annunciation messages are color coded to facilitate system identification and the priority of the alarm.

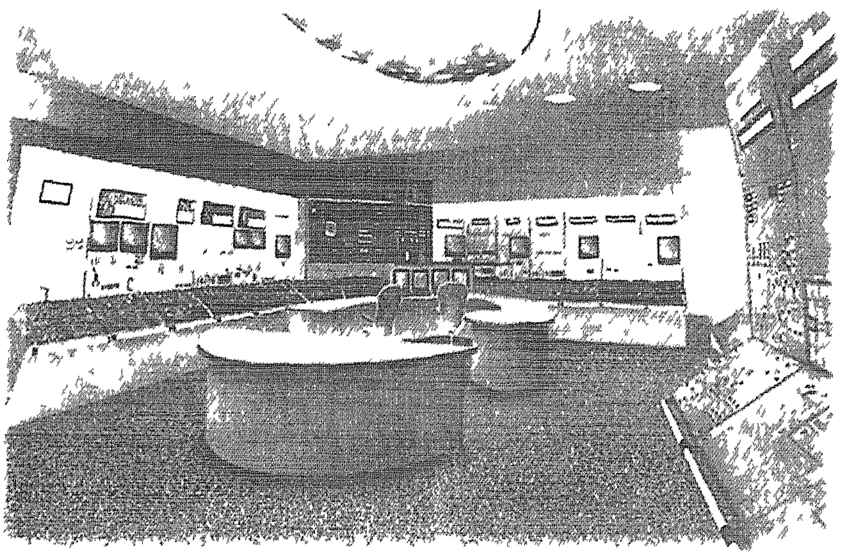

FIG 9 CANDU® 6 Control Room

6. SAFETY SYSTEMS

6.1 Design Principles

Special Safety Systems are incorporated in the plant for safety actions; namely reactor shutdown, decay heat removal and/or retention of released radioactivity. They are completely separate from the normal plant process systems, and perform no active function during normal operation They consist of:

- Shutdown System No. 1 (SDS1),
- Shutdown System No. 2 (SDS2),
- Emergency Core Cooling System (ECCS), and
- Containment System (containment structures, containment heat removal and isolation systems)

Each special safety system is designed to achieve a demand unavailability of less than 1/1000, demonstrated by on-line testing and the reactor may not be operated unless all are available. Systems to provide reliable services, such as electrical power, cooling water, and air supplies to the special safety systems are referred to as safety support systems. To guard against cross-linked and common mode events, all plant systems including the safety system, are assigned to one of two Groups (Group 1 and Group 2), and are separated spatially or by barriers.

6.2 Shutdown systems

The CANDU® 6 reactor incorporates two diverse, passive, shutdown systems that are independent of each other and from the reactor regulating system. Each shutdown system is capable of tripping the reactor and has sufficient negative reactivity to maintain it in a shutdown state. Each system is designed with an array of trip instrumentation so that, for any initiating event requiring shutdown, there are two different signals that trigger shutdown, for each system in turn.

Shutdown system No. 1 (SDS1) consists of mechanical shutdown rods that drop into the core upon a trip signal. The signal de-energizes the clutches holding the rods in place, releasing them into the reactor core.

Shutdown system No. 2 (SDS2) injects a concentrated solution of gadolinium nitrate into the low-pressure moderator to quickly render the core sub-critical. The injection is initiated by opening fast acting valves to pressurize poison tanks, one for each injection nozzle, with helium.

Each shutdown system has a set of trip parameters designed to protect against postulated system failures. The measurements for the trip parameters are triplicated, with a complete set of measurements provided in each of three channels for each shutdown system. Any measured parameter exceeding its set point will trip the corresponding channel, and any two tripped channels in the same shutdown system will shut down the reactor.

Independent neutron and process measurements provide signals to trip each shutdown system. Neutron measurements are obtained from self-powered in-core flux detectors and ion chambers. Process measurements include heat transport pressure, heat transport flow, reactor building pressure, steam generator low level, pressurizer low level, low steam generator feedline pressure, and moderator level. The reactor can be tripped manually in the main control room and in the secondary control area.

All information regarding trip parameters and the status and operation of the shutdown systems is displayed on dedicated panels in the main control room. Information for post-accident monitoring for operator action is also provided in the secondary control area.

6.3 Emergency core cooling (ECC) system

The ECC system is composed of three stages: high-pressure injection stage, intermediate stage, and recovery stage. The high-pressure injection stage uses pre-pressurized air to inject water into the HTS from water tanks. The intermediate stage supplies water from the dousing tank through duplicated redundant pumps. Once the water supply depletes, the recovery stage recovers water that collects in the reactor building sump and pumps it back into the HTS via the emergency cooling system heat exchangers using the same emergency cooling recovery pumps as in the intermediate stage.

The injection stage consists of one air gas tank and two water tanks. During normal reactor operation the gas tank operates at a pressure of 4.1 MPa, whereas the water tank pressure is slightly above atmospheric. The recovery pumps are two 100 per cent pumps. Each pump is supplied by Class III power and also by the emergency power supply system. The two 100 per cent heat exchangers in the recovery pump discharge line are designed to maintain the emergency cooling flow at about 50°C at entry to the heat transport system. Transfer between stages is fully automatic, requiring no operator action. Use of passive features is maximized to minimize the degree of complexity in valve stations, etc.

The ECC system is triggered automatically on a loss of coolant signal when the heat transport pressure drops to 5.5 MPa and one of the conditioning parameters (high reactor building pressure, high moderator level, high fuelling machine vault temperature, high steam generator room temperature) reaches its set point. The following actions take place:

- All gas isolation valves, high pressure injection valves, and D2O isolating valves are opened. This will break rupture discs in the injection lines and permit flow of high pressure water from injection tanks to all reactor headers of failed and unfailed loops.
- The main steam safety valves on the steam generators are opened to rapidly cool down the boilers and ensure a long-term heat sink, for small loss of coolant accidents.
- Valves in all lines interconnecting the two heat transport loops are closed. This will confine the consequences of a loss of coolant accident only to the loop containing the break.

The ECC dousing tank valves to the ECC recovery pumps suction will open automatically on a loss of coolant signal and one of the ECC pumps will automatically start when these valves are opened. If this pump fails to start (as indicated by a low pump discharge pressure), the standby ECC pump starts automatically.

As the dousing tank water depletes, the valves in the recovery line from the reactor building sump automatically open, the valves in the line from the dousing tank automatically close, and the cooling water valves open to supply service water to the ECC heat exchangers. The mixture of heat transport coolant and water from the high pressure and dousing tanks is pumped from the sump in the reactor building back to the heat transport system via the heat exchangers.

6.4 Containment

The containment system for accident protection consists of a post-tensioned concrete reactor building structure with an epoxy liner; atmosphere energy removal via an automatic dousing system and accident qualified building air coolers; access airlocks and an automatic containment isolation system.

The dousing tank is located in the dome of the reactor building and holds water for emergency dousing and emergency core cooling. About 500 cubic meters of water are reserved for emergency core cooling. The total capacity of the tank is about 2600 cubic meters. Dousing valves control the flow of water to six independent dousing spray header units located radially below the tank. Each spray unit has two butterfly valves in a downcomer between the tank and the spray header. The design dousing flow rate is about 4500 kg/s and any four of the six downcomers can provide this flowrate. With all six downcomers operating, the total spray flow is about 6800 kg/s.

A hydrogen mitigation system addresses the short and long-term release of hydrogen in the containment building. The hydrogen mitigation system maintains hydrogen in the containment building, at a level at or below ignition concentrations. Hydrogen is intentionally burnt as soon as the hydrogen-air mixture becomes flammable and before the hydrogen concentration reaches a detonation threshold.

Under normal operation conditions, the pressure within containment is slightly less than atmospheric and the containment ventilation dampers are all open.

The containment is automatically isolated in the event of a high radioactivity signal that may occur following a loss of coolant accident or any other event-releasing radioactivity into the reactor building. In the event of very small HTS leaks, the building coolers in the containment condense any steam discharged. The building pressure remains at atmospheric pressure and there may be some additional outflow of dried air through the ventilation system. In the event of larger breaks, the building pressure rises and at an overpressure of about 3.4 kPa, containment pressure sensors also initiate total containment closure. The containment pressure continues to rise and the dousing system begins to operate automatically at an overpressure of 13.8 kPa. Once the dousing system overtakes the pressure transient, the pressure begins to fall, and the building depressurizes to about atmospheric pressure by condensation on the building walls and cooling by the air coolers. Leakage of radioactivity from the isolated reactor building is restricted to very small levels. Continued heat removal from the atmosphere by environmentally qualified local air coolers ensures containment temperatures and pressures are controlled.

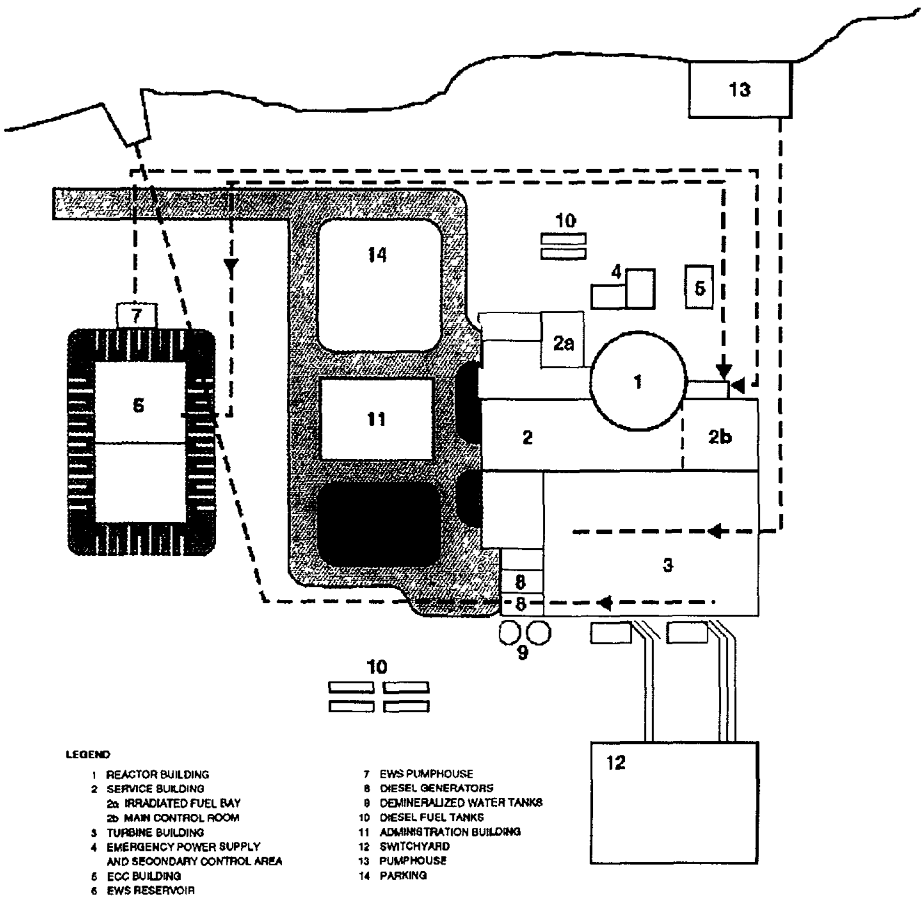

FIG. 10. Station Layout

7. SITE AND PLANT ARRANGEMENT

7.1 Typical site layout

CANDU® 6 is designed as a single-unit plant; multiple-unit stations use the single unit as a building block. The units of a multiple-unit station are self-sufficient, with all the facilities required for day-to-day operation. However, some support facilities, such as heavy water management and administration, may be integrated for more efficient and convenient operation.

A typical layout of a single-unit CANDU® 6 is presented in Figure 10. The principal structures include the reactor building, the service building and turbine building. Auxiliary structures include the pumphouse, secondary control area, the ECC building, the EWS pumphouse and the administration building.

7.2 Buildings and structure

7.2.1 Reactor Building

The reactor building houses the nuclear reactor and auxiliaries, primary heat transport system, fuel handling equipment, and instrumentation; major structural components are:

- the pre-stressed concrete containment structure,
- internal reinforced concrete structures, and
- the steel-lined reinforced concrete calandria vault.

The containment structure is separated from the internal structural systems. This provides flexibility in over-all building construction and no inter-dependence between the containment wall

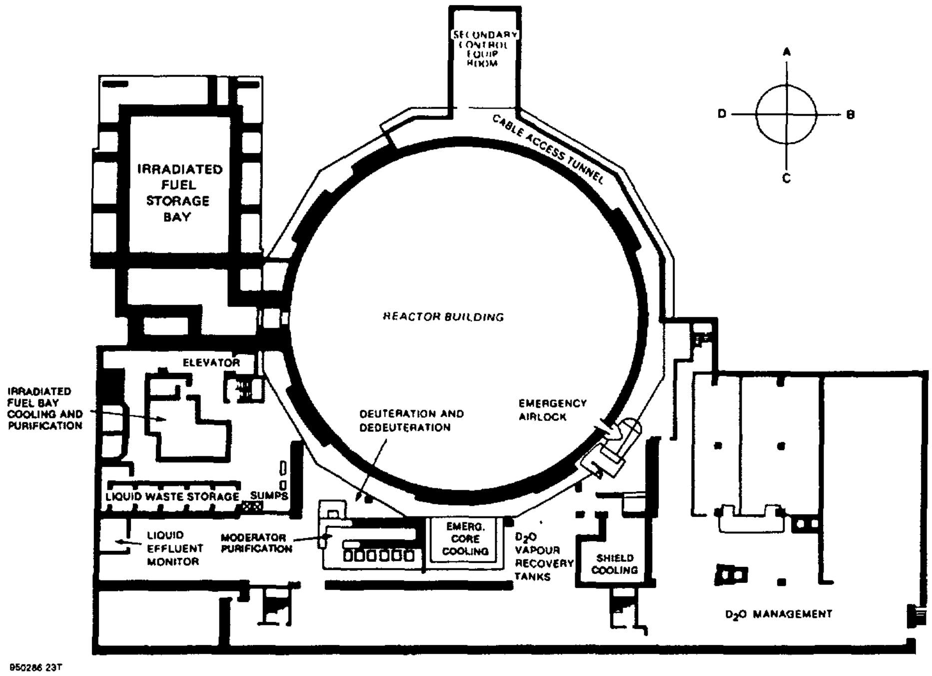

FIG. 11. Service Building Basement Plant - EL 93.90 m

and other structures. A large portion of the reactor building is accessible during operations, facilitating on-power maintenance, inspection and testing.

7.2.2 Service Building

The service building houses nuclear facilities including the control room, which can be located outside of the reactor building. More general service facilities, e.g., equipment maintenance shops and laboratory facilities, are located in the common area of this building.

The layout of rooms within the service building provides safety and efficiency for plant operation in terms of traffic patterns, radiation zoning and the routing of services between the buildings. The irradiated fuel storage facility is also in the service building. Figure 11 shows the basement plan of the service building, where major systems are located.

7.2.3 Turbine Building

The turbine building consists of a turbine hall, auxiliaries bay and two single-story annexes. Space is provided in the auxiliaries' bay for the electrical power distribution equipment. Water treatment plant and diesel generators are at grade level in the annexes.

Overhead traveling cranes in the turbine hall provide for erection and maintenance of the turbine generator and its auxiliaries. The turbine building is a reinforced concrete substructure, with a steel-framed superstructure, and steel roof trusses and insulated metal walls and roof.

7.2.4 Pumphouse

The pumphouse is a reinforced concrete substructure containing the condenser cooling water pumps, raw service water pumps, fire pumps, screens, and racks and screen wash pumps. A steel-framed superstructure provides houses the pump motors. Roof hatches allow installation and maintenance of the pumps.

7.2.5 Secondary Control Area (SCA) Building

This building contains the secondary control area, the emergency power supplies (diesel generators) and equipment to power the secondary control equipment to shutdown the reactor and maintain it in a shutdown state when the main control room is unavailable. The building has a one story steel superstructure and a reinforced basement.

7.2.6 Emergency Water Supply Pumphouse

The emergency water supply pumphouse, adjacent to the EWS reservoir, houses the fire-water pumps and emergency water supply pumps with their associated piping, valves and electrical equipment. The building is a reinforced concrete substructure consisting of a floor slab and foundation walls extending up to grade level and a one-story steel braced frame superstructure clad with sheet metal and built-up roofing.

7.2.7 ECC Building

The ECC building is on the northeast side of the reactor building and contains two ECC system water tanks and the ECC system air tank. The substructure is a reinforced concrete open top box with the base slab below and walls terminated above grade. The superstructure is a one story steel braced frame construction. For access to the basement, steel platforms and stairs are provided. The building is clad with insulated metal siding and built-up roofing.

8. HEAVY WATER MANAGEMENT

Heavy water management consists of systems and facilities to clean-up, upgrade, store, collect and supply make-up of heavy water for the moderator and heat transport systems. The station design prevents the loss of D_2O from the reactor systems. Special measures recover and upgrade D_2O that escapes. A D_2O upgrader facility is integrated into the station design.

The heavy water cleanup system removes dissolved particulate and organic impurities from heavy water recovered from process systems and has a product suitable for upgrading in the heavy water management facility. The heavy water management facilities are in the service building. Provisions ensuring optimum D_2O management are:

- extensive use of welded joints, with the number of mechanical joints in heavy water systems kept to a minimum,
- heavy water and light water systems are segregated as much as possible,
- a D_2O liquid recovery system is provided,
- the building containing most heavy water systems is sealed and has a minimum through ventilation flow,
- air entering and leaving the reactor building is dried to minimize D_2O downgrading and loss, and

- dry air is maintained within the building by closed circulation and drying systems so increases in humidity can be readily detected. Heavy water vapor removed by the dryers is recovered and upgraded.

Part of the CANDU® 6 program is simplifying and improving routing of ventilation systems and D_2O vapor recovery flows, to cut heavy water losses, already exceptionally low, by more than 50 per cent.

9. RADIOACTIVE WASTE MANAGEMENT

Radioactive waste management systems permit on-site collection, processing and handling of all radioactive wastes produced. Facilities provide for either interim site storage or disposal on-site or off-site.

9.1 Liquid Radioactive Waste Management System

The liquid radioactive waste management system receives all active liquid wastes and provides storage, sampling, treatment and dispersal. The system consists of concrete storage tanks (for low activity and for normal activity wastes). Filters and an ion exchanger reduce radioactivity as required. The liquid waste management systems are in the nuclear steam plant services portion of the station services building.

9.2 Gaseous Radioactive Waste Management System

The gaseous radioactive waste management system handles potentially active airborne discharges from the reactor building, irradiated fuel storage bay area, decontamination center, heavy water handling area, and active ventilation exhausts. All active or potentially active gases, vapors, or airborne particles are monitored and filtered, if necessary, prior to release via the ventilation exhaust duct or stack. Active gases vented from the heat transport system are released to the active ventilation system only after removal of heavy water and a delay to permit decay of short-half-life isotopes. The stack effluent is continuously sampled to detect the presence of tritium, particulates, iodine-131 and noble gases.

9.3 Solid Radioactive Wastes

Solid radioactive wastes include a variety of solid materials that may be contaminated. These include filters, ion exchange resins, activated components and common materials such as paper, plastic, wood etc. A very small volume of radioactive liquid waste that can not be handled in the radioactive liquid waste management systems may be included in the solid wastes. These and some organic solvents, such as scintillation solutions, are immobilized using an inert absorbent. The solid radioactive waste area is normally established on the reactor site, some distance from the reactor building. It typically comprises an elevated earth-fill platform with an approach ramp for vehicles, enclosed by a security fence, with concrete waste storage structures embedded in the earth of the platform.

9.4 Dry Spent Fuel Storage

The CANDU® 6 spent fuel bay capacity with the most up-to-date storage rack design can store up to 20 years of spent fuel; additional storage space is required for 40-year operations. Expanding spent fuel cooling pools in nuclear power plants generally is logistically difficult and expensive, dry storage systems evolved out of this need. Dry storage for fuel bundles is relatively simple, and has been proven in dry storage units in operation at several units.

The CANDU® MACSTOR dry spent fuel storage system is an air-cooled concrete module housing a number of metal canisters containing spent fuel. This provides highly efficient heat rejection, excellent shielding and complete structural soundness. Figure 12 illustrates this system. Spent fuel handling is simplified by transferring fuel from the spent fuel bay in the same baskets, into dry storage modules. Dry spent fuel storage can be applied as soon as after 6 years pool storage, and is licensed locally.

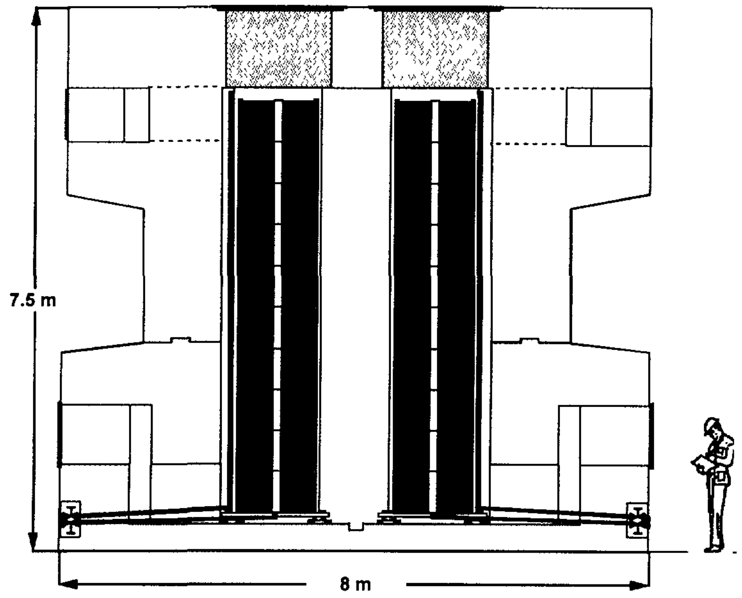

FIG. 12. Cutaway of a MACSTOR Module

10. TECHNICAL DATA

These data are typical values for a CANDU® 6 nuclear power plant. Actual values for a specific plant may vary depending on site conditions.

<u>General Plant Data</u>		
Design life	40	Yrs
Power plant output, gross	715	MWe
Power plant output, net	665	MWe
Reactor thermal output	2064	MW
Power plant efficiency, net	30.8	%
Cooling water temperature	18	°C
<u>Nuclear Steam Supply System</u>		
Number of coolant loops	2	
Primary circuit volume	220	m^3
Steam flow rate at nominal conditions	1033	kg/s
Feedwater flow rate at nominal conditions	1034	kg/s
Steam temperature	260	°C
Steam pressure (gauge)	4.7	MPa
Feedwater Temperature	186.7	°C
<u>Reactor Coolant System</u>		
Primary coolant flow rate	7700	kg/s
Reactor outlet header operating pressure	9.9	MPa
Reactor inlet header operating temperature	260	°C
Specific enthalpy change across core (aver.)	271.5	kJ/kg
Reactor outlet header operating temperature	310	°C
D_2O inventory	465.5	Mg

Reactor Core

Active core length	5.94	m
Core radius, effective	3.38	m
Number of lattice Cells	380	
Cell array	square	
Lattice pitch	285.75	mm
Fuel inventory	95	t U
Fuel material	Sintered U_2O	
Fuel assembly(fuel bundle) total length	495	mm
Fuel design (rod array)	Circular array	
Number of fuel bundles	4560	
Number of fuel rods (pins) /assembly	37-element	
Average discharge burnup of fuel	7000	MWd/t
Cladding tube material	zircaloy-4	
Cladding tube wall thickness	0.42	mm
Outer diameter of fuel rods	13.08	mm
Overall weight of fuel bundle	23.5	kg
Number of shutdown/control units - vertical	59	
Number of shutdown units - horizontal	6	
Reactivity control material - out of core	stainless steel	
- in core	zircaloy/stainless steel/cadmium	

Reactor Vessel

Calandria shield tank assembly (CSTA)(diam.)	7.65	m
Wall thickness of cylindrical shell – calandria	28.6	mm
- shield tank	25.4	mm
CSTA overall length	7.82	m
Base material – calandria	304L Stainless Steel	
Base material – shield tank	carbon steel	

Fuel Channels

Channel length	10.82	m
Pressure tube length	6.30	m
Pressure tube inside diameter	103.4	mm
Number of pressure tubes	380	
Flow rate single channel, nominal	117	m^3/hr
Est. pressure drop across 12 bundles	758	kPa
Max. channel power (nominal)	6.46	MW

Steam Generators

Steam generator type	vertical U-tube with integral preheaters	
No. of steam generators	4	
Tube dimensions (outer dia./thickness)	16/1.12	mm
Shell and tube sheet material	carbon steel	
Tube material	Incoloy 800	

Reactor Coolant Pump

Number	4	
Design pressure (gauge)	12.9	MPa
Design temperature	279.4	°C
Design flow rate(at operating conditions)	2.228	m^3/s
Operating temperature	266	°C
Pump head	215	m
Power demand at coupling, hot	6.7	MWe
Pump casing material	carbon steel	

Pressurizer

Total volume	45.3	m^3
Design pressure (gauge)	11	Mpa
Design temperature	310	°C
Heating power of heater rods	5 x 200	kWe
Material	carbon steel	

Primary Containment		
Containment type	reinforced prestressed concrete	
Overall form (spherical/cyl.)	cylindrical	
Inside diameter	41.45	m
Wall thickness	1.07	m
Height (top of base slab to top of dome)	51.21	m
Free volume	4 8 477	m^3
Design pressure (LOCA) (gauge)	124.1	kPa
Reactor Auxiliary Systems		
Reactor water cleanup - (purification)	24	kg/s
Reactor water cleanup - filter type	disposable cartridge	
Purification heat exchanger capacity	50	MW(th)
Shutdown cooling heat capacity	13.3	MW(th)
Reactivity Control		
Shut-off units (SDS1)		
Quantity	28	
Type	stainless steel clad cadium tube	
Static reactivity worth (28 rods)	-80	milli-k
Liquid Injection System(SDS2)		
Quantity	6	
Type	horizontal nozzle inject liquid poison into moderator	
Poison	Gadolinium nitrate	
Reactivity worth, long term	-300	milli-k
Liquid Zone Control Units		
Quantity	6	
Type	vertical tubes divided into compartments that can be filled with light water	
Reactivity worth	-7.5	milli-k
Mechanical Control Absorber Units		
Quantity	4	
Type	stainless steel-clad cadium tube	
Reactivity worth	-10	milli-k
Adjuster Units		
Quantity	21	
Type	variable thickness tube plus center rod	
Reactivity worth	-15	milli-k
Power Supply Systems		
Main output transformer -rated voltage	380/20	kV
Main output transformer -rated capacity	812/3	MVA
Unit service transformer - rated voltage	20/13.8/13.8	kV
Unit service transformer - rated capacity	59.4/79.2	MVA
System service transformer -rated voltage	380/13.8/13.8	kV
System service transformer -rated capacity	59.4/79.2	MVA
Standby diesel generating units		
- Group 1 (non-safety) number/rated power	2 / 6.5	MWe
- Group 2 (safety) - number/rated power	2 /1.36	MWe
Turbine Plant		
Number of turbines per reactor	1	
Type of turbine	Tandem Compound	
Number of turbine sections per unit	HP/LP/LP/LP	
Turbine speed	1 800	rpm

Generator		
Type	4 pole H_2 cooled	
Rated power	856	MVA
Active power	720	MWe
Voltage output	20	kV

11. PROJECT STATUS AND PLANNED SCHEDULE

The CANDU® 6 plant design is a "living" design, which has undergone continuous evolution from the time of the original units that entered service in 1983. Since then, the design has been maintained project-ready, while individual improvements have been incorporated as applicable and as mandated by each customer. The present enhancement program was started in 1996, and will include successive improvements when fully proven and pre-engineered. In this way, the design is up-to-date and project- ready. The program objective is to have short-term design enhancements completed by the year 2000. Continued design improvements will be made to adapt to operational and other experiential feedback. In the longer term, this program will lead to adaptations to the medium-sized CANDU® 6 concept to lead to an Advanced CANDU® project.

Implementation of a CANDU® 6 project is proven achievable within very efficient project schedules. The Wolsong 2 project was completed in 1997 within a 77-month total schedule (contract effective to commercial operation). The follow-on unit, Wolsong 3, was completed in 1998 in an even more demanding 69-month total project schedule. The recently-started Qinshan 2-unit project, the first CANDU® in China, is on a new untouched site within the Qinshan nuclear park and will see first unit completion in 72 months (17 months engineering plus 55 months construction and commissioning). Since the start of the Qinshan units, further construction studies have shown that a shorter schedule can be achieved with the application of small-scale pre-fabricated modules, and with the further use of open-top construction. A development plan has been defined which will complete the development of a "reference" 66 -month total project schedule (contract effective to in-service).

REFERENCES

[1] AECL CANDU® 6 Technical Outline, Revision 1, November 1996.

[2] AECL CANDU® Technical Summary by Raph S. Hart, revised October 1997.

[3] AECL CANDU® 6 Technical Description, January 1998.

IAEA-SM-353/50

CANDU 9 DESIGN

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Abstract

AECL has made significant design improvements in the latest CANDU nuclear power plant (NPP) - the CANDU 9. The CANDU 9 operates with the energy efficient heavy water moderated reactor and natural uranium fuel and utilizes proven technology. The CANDU 9 NPP design is similar to the world leading CANDU 6 but is based upon the single unit adaptation of the 900MWe class reactors currently operating in Canada as in integrated four-unit configurations. The evolution of the CANDU® family of heavy water reactors (HWR) is based on a continuous product improvement approach. Proven equipment and systems from operating stations are standardized and used in new products. As a result of the flexibility of the technology, evolution of the current design will ensure that any new requirements can be met, and there is no need to change the basic concept. This paper will provide an overview for some of the key features of the CANDU 9 NPP such as nuclear systems and equipment, advanced control and computer systems, safety design and protection features, and plant layout. The safety enhancements and operability improvements implemented in this design are described and some of the advantages that can be expected by the operating utility are highlighted.

1. Introduction

Building on the success of the 4 unit station at Bruce B which began commercial operation in 1980s, four additional 900 MW(e) class units were commissioned at Darlington in the early 1990s. The CANDU 9 is a 935 MW(e) reactor based on the multi-unit Darlington and Bruce B designs with some additional enhancements from our ongoing engineering and research programs [1,2,3]. The CANDU 9 reactor continues the evolutionary improvement approach adopted for the CANDU 6. In addition to the advantages of using proven systems and components, CANDU 9 offers improvement features with enhanced safety, a control centre with better operability and improved project delivery in both engineering and construction.

Enhanced competitiveness of the CANDU product is assured by incorporating improvements based on updated technologies, including safety technology, the rapidly advancing information technology and modern construction methodology. Reduced project implementation risk for CANDU 9 is assured by up-front engineering and licensing prior to contract start.

A cutaway view of the CANDU 9 plant is shown in Figure 1-1.

The principal CANDU 9 requirements include:

a. A CANDU plant with a gross electrical output of over 935 MWe, utilising proven systems components and concepts.

b. Reduce specific capital cost, construction schedule by minimizing equipment installation time, and unit energy cost.

c. The enhancement of traditional CANDU advantages including real safety, low radiation exposure, high capacity factor, ease of maintenance, and low operating cost.

d. Specific design targets include a reduction in occupational doses to below 1 person Sv per year and a 90 percent or greater lifetime capacity factor.

e. Human factors considerations in the design of systems, facilities, equipment and procedures. Human factors driven design considerations are applied plant–wide consistently where there is an interface with plant personnel.

f. The plant component shall be designed to operate for up to 60 years with allowance for expeditious replacement. The provision for "easy" replacement means quick, simple and without complex

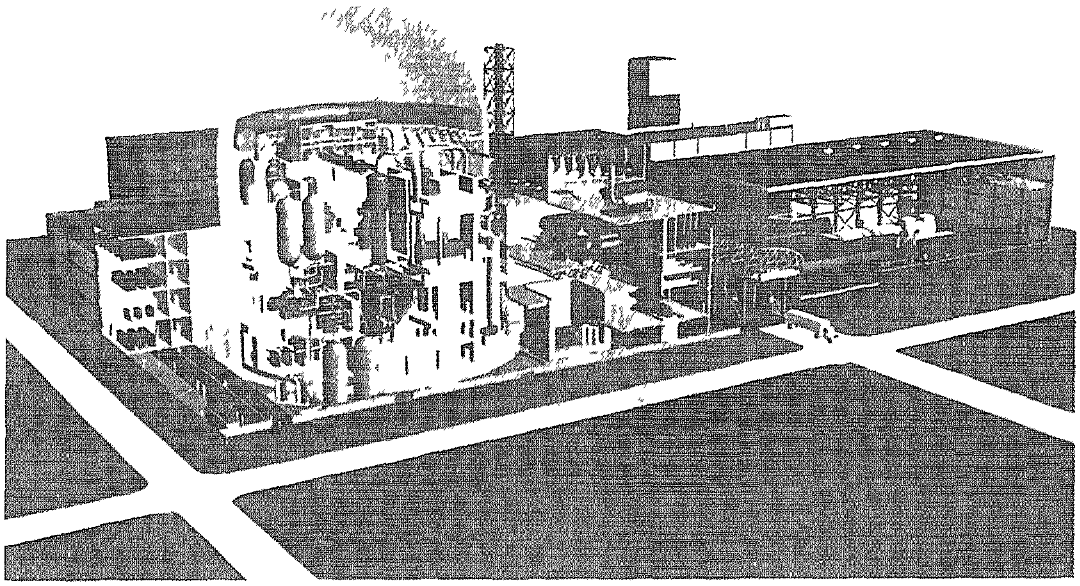

Figure 1-1. Cutaway view of CANDU 9 plant.

tooling, thereby minimising radiation exposure. Such work may be necessary to extend operating life after more than 30 years of operation for pressure tubes, and more than 40 years for replaceable equipment such as steam generators.

g. Provision for improved plant capability from a maintenance/in–service inspection viewpoint

   1. a total of 28 days over a two year period of station operation required for scheduled maintenance outages, and
   2. accommodation of any major maintenance outages that may become necessary, including large component replacement, major systems modernisation, or equipment refurbishing, lasting up to 180 days every 10 years.

2. Description of the nuclear systems

2.1 Primary circuit

The coolant flow in the CANDU 9 heat transport system is in the "figure–of–eight" pattern employed in all CANDU reactors, with the heat transport pumps in series and the coolant making two core passes per cycle. The equipment arrangement results in bi–directional coolant flow through the core

Improvements were made to the CANDU 9 heat transport system (HTS), relative to the reference configuration at Bruce B by interlacing the feeders so that adjacent channels are alternatively connected to separate inlet and outlet headers. In this way the fuel channels served by each inlet header (25% of total) are uniformly distributed throughout the core. This arrangement minimises the positive reactivity insertion from a large pipe break in the HTS.

Another improvement is the provision of a larger pressurizer capable of accommodating changes in volume of the reactor coolant in the HTS from full power to shutdown condition at 100°C. This additional inventory will allow the heat transport system to remain filled with water to enhance stable thermosyphoning after events such as a loss of forced circulation, a loss of main heat sinks, or a spurious cooldown.

2.2 Reactor core and systems

The design and the neutronic characterics of the CANDU 9 reactor core closely follows that of operating CANDU reactors. The core incorporates the standard CANDU geometrical arrangement of horizontal fuel channels in a square lattice.

Standard CANDU fuel consists of 37 elements of uranium dioxide sheathed in Zircaloy and held together as a bundle by end–plates. There are 12 fuel bundles in each fuel channel.

The reactor fuel is changed on a routine basis with the reactor operating at full load using two fuelling machines, one located at each end of the reactor. Each CANDU 9 fuelling machine is mounted on a fuelling machine carriage, which travels on a track that runs between the fuel ports at the containment building wall and the fuelling machine vault.

On–power fuel changing is normally performed under automatic remote control. One fuelling machine clamps on to the new fuel port and accepts new fuel bundles. The machine is then connected to a fuel channel for refuelling. The other fuelling machine, having discharged all irradiated fuel bundles to an irradiated fuel bay via the irradiated fuel port, similarly moves to the opposite end of the fuel channel selected for refuelling.

Space, handling and lifting facilities are provided in the irradiated fuel bay areas for transferring irradiated fuel to transport equipment for shipping to on–site dry fuel storage facilities.

2 3 Primary components

Reactor Assembly

The reactor vessel consists of a cylindrical calandria (Figure 2-1), and end shield assembly, enclosed and supported by the cylindrical shield tank and its end walls. The calandria shell is closed and supported by the end shields at each end, filled with heavy water moderator. The shield tank is filled with

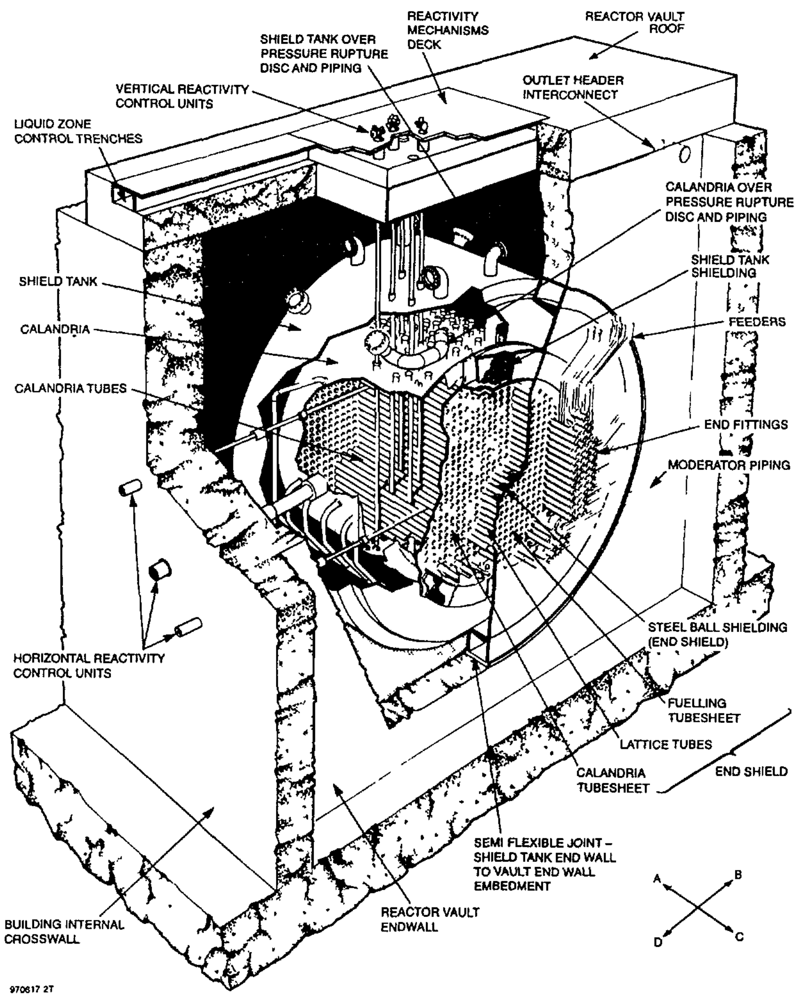

Figure 2-1. Reactor assembly.

light water and the spaces between the two tubesheets of both end shields are filled with steel balls and water. This shielding allows personnel access to the reactor face during reactor shutdown. The effect of integrated flux, or fluence, on the calandria assembly for operation up to 60 years has been included in the reactor assembly design.

Fuel Channels

The horizontal fuel channel assemblies are of the same design as those of CANDU 6. Each pressure tube is thermally insulated from the low temperature, low pressure moderator by the carbon dioxide filled gas annulus formed between the pressure tube and the calandria tube. Provision is made to promptly detect leakage from the moderator or heat transport system into the annulus of any fuel channel by continuously monitoring the moisture content of the gas in the annulus gas system.

The pressure tubes are made of a cold–worked, zirconium–2.5% niobium alloy, which offers high strength, low neutron absorption and high corrosion resistance. The pressure tubes which are the only components in a CANDU reactor subjected to a combination of high radiation, high stress, and high temperature, have a minimum service life of 30 years and are readily replaceable.

Steam generators

The CANDU 9 steam generators, which are of the same design as those of Darlington, consist of an inverted vertical U–tube bundle in a cylindrical shell.

High recirculation ratios and relatively low heat flux, in combination with comprehensive chemistry control, careful material specification and detailed attention to design assure long life and low maintenance requirements for CANDU steam generators. An all-welded primary head divider plate is incorporated to minimize leakage during normal operation. For CANDU 9, additional inspection ports are added near the tubesheet and at tube support plates on the secondary side to provide increased access for cleaning, inspection and water lancing.

Reactor coolant pumps

The CANDU 9 heat transport pumps are of the same design as those of the Bruce B and Darlington Nuclear Generating Stations. Each pump is driven by a vertical, totally enclosed, air–water cooled squirrel cage induction motor. The pump seals and bearings can be removed without removing the motor. The pump shaft sealing arrangement consists of three mechanical seals and one back–up seal in series. Each mechanical seal is designed to withstand the full differential pressure.

2.4 Reactor auxiliary systems

Moderator System

The heavy water moderator in the calandria is used to thermalize fast neutrons produced by fission and is circulated through the calandria and moderator heat exchangers to remove the heat generated in the moderator during reactor operation.

The moderator system is fully independent of the heat transport system. The moderator system includes two pumps and four plate-type heat exchangers. The heavy water in the calandria functions as a heat sink in the unlikely event of a loss of coolant accident coincident with failure of emergency core cooling. The capability of this heat sink is assured by controlling the heavy water temperature in the calandria within specified limits and providing the means for inventory make–up from the reserve water system.

Shutdown Cooling System

After shutdown and during maintenance, the shutdown cooling system removes decay heat and cools the heat transport system to a temperature suitable for maintenance of the heat transport and auxiliary systems components. For CANDU 9, design specifications for the system and components are enhanced so that the system can be placed into service under zero power, full pressure, hot conditions (265 deg C) to cooldown the heat transport system, and is therefore a backup to the steam generators for emergencies. This additional capability enhances the overall heat sink reliability for CANDU 9.

Reserve Water System

The reserve water tank provides an emergency water supply for emergency core cooling, backup feedwater supply as well as providing a make-up source for the shield tank, moderator and heat transport systems.

The reserve water tank is also connected to the normal end shield cooling circuit. During normal reactor operation, the reserve water tank acts as the head tank for the end shield cooling circulation pumps. However, in the event of process failures such as a loss of forced circulation in the end shield cooling circuit or a loss of cooling water to the end shield heat exchanger, the reserve water tank with its large water inventory acts as a passive heat sink. The layout of the equipment and the piping connection between the end shield of the reactor core and the reserve water tank are designed to facilitate enhanced thermosyphoning for heat removal from the end shield. [4]

Steam and Feedwater Systems

Steam produced in the steam generators is fed by separate steam mains to the turbine steam chest. The steam pressure is normally controlled to a constant value by varying reactor power to match the turbine–generator demand.

The Group 1 feedwater system comprises three 50 percent capacity main feedwater pumps on Class IV power that take suction from the deaerator storage tank, and a diesel driven auxiliary feedwater pump. Feedwater from the regenerative feed heating system is supplied to the steam generators through two separate feedwater mains.

A Group 2 Feedwater System provides emergency water to the steam generators automatically for decay heat removal for approximately 10 hours, providing back-up to the Group 1 Feedwater Systems. This new system is seismically qualified and can operate at full steam generator pressure so as to cope with all the possible operating conditions in the steam generators. The available operator response time, generally 8 hours, has been attained for accident conditions requiring steam generator heat sinks.

HVAC

The reactor building cooling system controls air temperatures in both accessible and inaccessible areas of the reactor building during reactor operation. The system also remains functional during a loss–of–coolant accident or a steam main failure. The system consists of vault cooling units and local cooling units for the shutdown cooling and moderator rooms and the accessible areas. Each cooling unit comprises air coolers, dampers and fans as well as piping and instrumentation.

The reactor building ventilation system provides air exchange and air distribution, maintains the reactor building at a slightly lower than atmospheric pressure, and provides filtration of activity.

2.5 Operating characteristics

CANDU 9 can operate continuously in the reactor–following–turbine mode and is capable of load following cycles that typically involve a rapid reduction of power from 100% to 60%, steady–state operation at 60% power for 6 hours, and a return to near full power over the following four hour period.

For reactor power increases, the nuclear steam plant portion of the plant is capable of manoeuvring at the following rates:

Power Range	Maximum Rate
0 - 20 percent of full power	4 percent of actual power per second
20 - 80 percent of full power	0.8 percent of full power per second
80 - 100 percent of full power	0.17 percent of full power per second

The overall plant power manoeuvring rate is a function of turbine design, and is typically limited to 5 to 10 percent of full power per minute.

The turbine bypass system to the condenser is capable of accepting the entire steam flow during a reactor power setback following loss of line or turbine trip, thereby avoiding any steam discharge to the atmosphere. The steam flow is initially 100 percent, but decreases to a steady state value in the range of 60 percent after several minutes.

3. Description of turbine generator plant system

The turbine–generator, feedwater and condensate plant is of conventional design. Requirements are specified, however, to assure performance and integrity of the nuclear steam plant. These include, for example, materials requirements (titanium condenser, absence of copper alloys in the feed train), feed train reliability requirements, feedwater inventory requirements and turbine bypass capability.

4. Instrumentation and control systems

4.1 Control centre design and operational interface

The design strategy for the CANDU 9 advanced control centre is to build upon the proven accomplishments of the control centre features of operating CANDU stations, and to improve operational tasks based on operations feedback, with a systematic and rigorous design process. The design focus for advanced features [5] is to improve the operability of the station, decrease the likelihood of operator or maintainer errors and to facilitate the achievement of higher production capacity factors while maintaining safety standards and providing improved maintenance/diagnostic capabilities.

Human factors principles are used in the design of the operator interfacing functions. The CANDU 9 control centre is consistent with current and emerging Canadian Standards Association (CSA) standards and with accepted international practices for nuclear plant control room design. Consideration of emergency operating procedures, critical safety parameters and post accident management standards are included in the control centre design process. Verification and validation activities ensure a functional control centre human system interface.

The main control room contains the main operator console, the fuel handling console, the safety system console, a shift interrogation console, and the main control room panels (Figure 4-1).

From the main operator console, the operator can perform supervisory and device control of the plant as it manoeuvres from zero power cold to full power. An annunciation interrogation workstation is provided at the main operator console. The consoles are arranged to give the operator an unobstructed view of alarm message video display units and central overview display. The main operator console has function based keyboards which allow interactive control and display commands.

The operator has comprehensive access to current and recorded data on all plant systems. Alarm annunciation follows existing CANDU practice, with improvements in the interactive use of "alarm filters", priority and system classification, etc. Automatic identification of the entry conditions to key procedures, enhanced alarm status review features, and the creation of high level alarms to aid operator diagnosis of plant fault conditions, are major improvements over past practice.

From the safety systems console the operator performs (computer-aided) testing and monitoring of the four special safety systems. Computer based displays available for the safety systems follow the plant conventions for video display unit interaction and format. Seismically qualified cabinets and instrumentation are used for all special safety systems in the main control room. The CANDU 9 main control room structure is seismically qualified for a design basis earthquake.

The human-system interface in the secondary control area is a duplication of the corresponding Group 2 control locations, layouts and capabilities present in the main control room, except safety system testing. This will ease the operator's task in relocating from one control area to the other. The design basis requires that the operator leaves for the secondary control area if the main control room becomes uninhabitable.

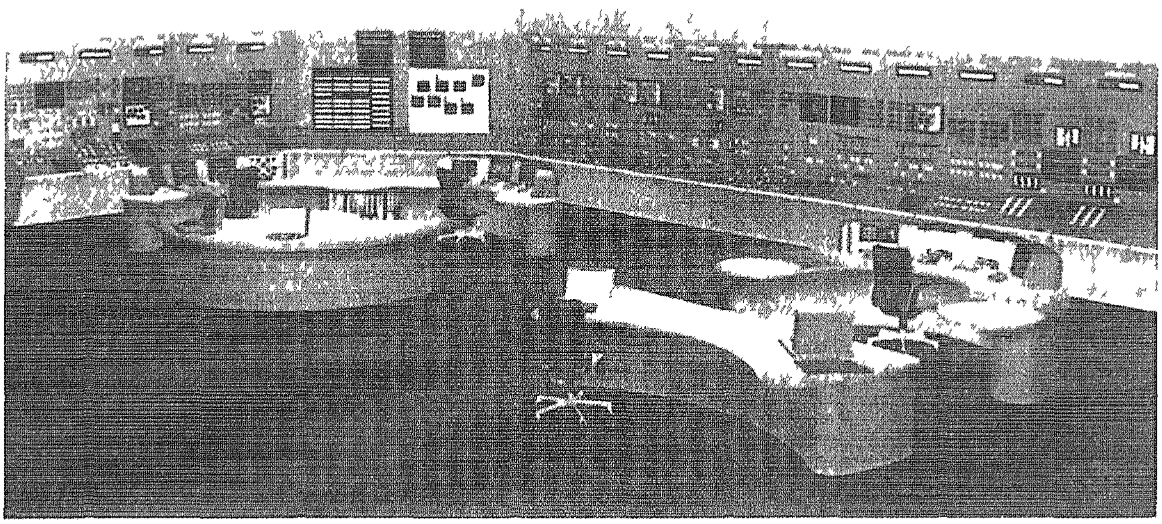

Figure 4-1. Main control room

The control centre layout makes space provision for a technical support centre. For utilities which are required to comply in specific detail with NRC NUREG 696, this facility provides for the required staff complement and safety parameter display facilities and related emergency support documentation.

Distributed Control System

Most of the Group 1 control functions are implemented by the Distributed Control System. The Distributed Control System also provides some control related data acquisition for the monitoring, alarm annunciation, display and data recording functions performed by the Plant Display System. The Distributed Control System also receives and executes operator commands entered via the Plant Display System.

The Distributed Control System is a modular digital control system which uses a number of programmable digital controllers connected to data highways. The data highways data transmission method provides very high data security. The system includes comprehensive fault detection, redundancy, and switchover features, to provide a very high degree of immunity to random component failures.

All control functions are implemented by programs in small powerful processor modules which are used in redundant pairs. The processors are programmed using control function block diagrams.

The distributed control system is partitioned into five independent functional segments, one for each of the major groups of plant systems. This functional partitioning provides a defence against common-mode faults (Figure 4-2).

Plant Display System and Safety System Monitor

The plant display system provides the monitoring, supervisory control and data handling facilities for the normal power production systems and also has the ability to monitor the Group 1 special safety systems and Group 2 systems. Some control related Group 1 information from the plant is fed to the plant display system via the distributed control system in addition to the facilities existing to receive information from other devices not involved in the distributed control system operation. This provides a complete plant status database of the normal power production systems information accessible to the operator. Operator setpoints and commands for the control functions are input to the distributed control system via the plant display system.

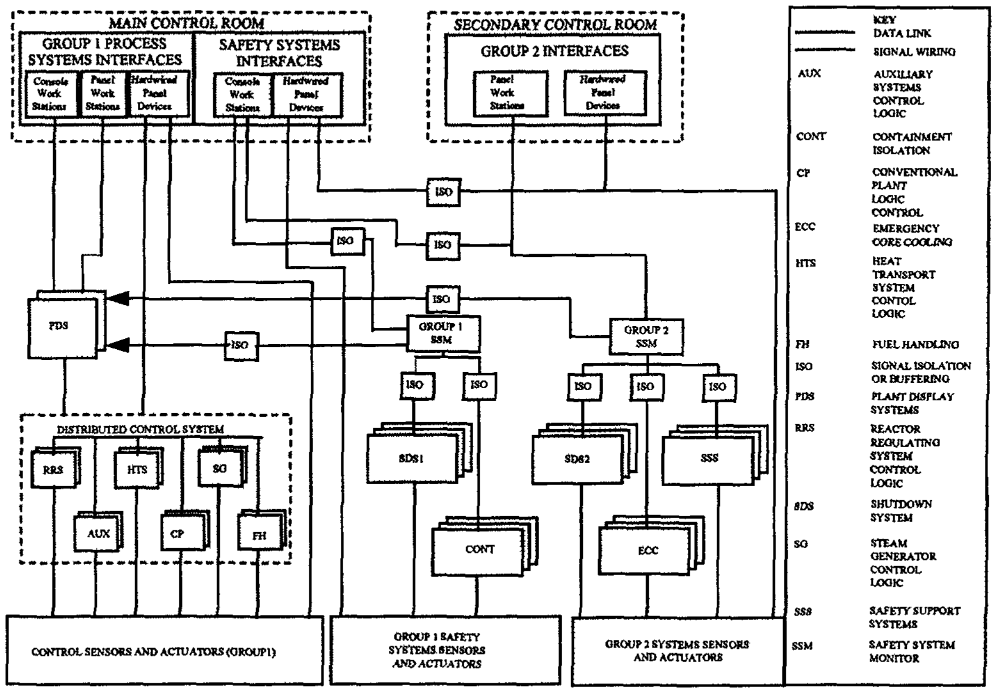

Figure 4-2. CANDU 9 instrumentation and control overview.

The two safety system monitors provide the monitoring, testing and data handling for the Group 1 special safety system and for the Group 2 systems. The information is fed to the safety system monitors from the channellized computers in the individual special safety systems.

The plant display system and the safety system monitors are the logical evolution of the successful application of digital computers in other CANDU plants. Each is highly redundant and modularised so as to maintain function availability in excess of 99 percent. High availability is achieved by ensuring that each of the major subsystems is either duplicated or provided with modular redundancy. All subsystems except for the graphic operator stations are duplicated. The number and arrangement of operator stations provides sufficient system redundancy. With their open architecture it is possible to expand the systems to meet future requirements.

CANDU 9 Post Accident monitoring is a conventional hardwired system made of discrete panel mounted devices located in both the main control room and the secondary control area. All PAM information will be clearly and uniquely distinguishable from other systems.

4.2 Power supply systems

The station service distribution supplies process and instrumentation loads within the plant. Group 1 provides power to production related equipment and to safety support systems. Group 2 is seismically and environmentally qualified and supplies power to all special safety systems and safety support systems, and to the secondary control area.

Standby power is supplied from both Group 1 as well as Group 2 for high reliability. The Group 1 Class III standby power supply system is provided by two diesel–generator units. The Class III shutdown loads are duplicated, one complete system being fed from each diesel–generator. There are two additional diesel–generator sets providing Group 2 Class III emergency power supply system. Each of these diesel-generators can supply the total safe shutdown load of the unit.

In the event of failure of the normal Class IV power sources, all four standby generators start automatically. The diesel–generators come up to speed and accept key loads within 30 seconds, and full load within two minutes. The fuel system has the capacity to supply the diesel generators for seven days of continuous operation at nominal load.

The reactor protective and safety systems, control logic, instrumentation, computers, critical motor loads, essential and emergency lighting and switchgear operation are supplied with uninterruptible power. The uninterruptible power supplies provide AC and DC power as required to the Class II and Class I systems. The Group 1 batteries are sized to support all the loads connected for up to 60 minutes following an interruption to the normal (Class III) source. AC power is obtained through static inverters. The Group 2 batteries are utilised for the emergency generators control logic and switchgear operation, and power the safety systems detection and initiation functions.

5. Safety Design

5.1 Safety requirements and design philosophy

CANDU design practice places emphasis on both inherent and engineered safety features to ensure that the plant can be safely operated and can respond to transients and accidents without causing undue risk to the plant personnel, the public and the environment.

Furthermore, accidents must be shown to have acceptable consequences, not only when the safety systems work, but also if a safety system is unavailable or impaired. This overall safety approach is achievable because there are at least two independent ways of providing the safety functions of shutdown and decay heat removal.

The concept of grouping and separation of safety related systems has been an integral component of CANDU plant designs for many years. For hazards such as earthquakes, fires, floods, missiles, etc., the plant is protected through implementation of a two-group approach. All plant systems are assigned to one of two Groups (Group 1 or Group 2). The systems of each Group are capable of shutting down the reactor, maintaining cooling of the fuel and providing plant monitoring capability, even if all of the other Group of systems is unavailable. For the CANDU 9 design, this concept has been enhanced through additional redundancy and diversity in the provision of cooling water, power supplies, and plant monitoring devices to maintain the controlled shutdown and cooldown condition. In particular, separate, seismically qualified Group 2 cooling water system, Group 2 power distribution system and Group 2 feed water system have been provided to assure heat removal after common mode events including an earthquake.

5.2 Safety systems features

The primary engineered safety features include the special safety systems: two shutdown systems (SDS1 and SDS2), the emergency core cooling system (ECCS) and the containment system. These systems have the dedicated role of mitigating the effects of postulated accidents, and are designed to be:

- Independent,
- Separated physically,
- Redundant,
- Testable during service, to meet a reliability target of 999 times out of 1000 tries, and
- Diverse in design and operation.

Shutdown Systems

The CANDU 9 reactor incorporates two diverse, passive, shutdown systems which are independent of each other and from the reactor regulating system.

Shutdown system No. 1 (SDS1) consists of mechanical shutdown rods.

Shutdown system No. 2 (SDS2) injects a concentrated solution of gadolinium nitrate into the low pressure moderator to quickly render the core subcritical.

The trip decision logic for each shutdown system is implemented through triplicated digital "trip computers" using rigorous software development methodologies and mathematical software verification techniques.

A computerized monitoring and test system in CANDU 9 provides the operator with indications of all shutdown system parameters and assists the operator in testing. The system prompts the operator, executes the testing, and records the test results. The test frequency depends on the unavailability requirement and the equipment failure rate for each trip variable. A test is automatically terminated if another trip channel goes into a tripped state.

Emergency Core Cooling System

The emergency core cooling system (ECCS) supplies light water coolant to the reactor and maintains fuel cooling in the event of a loss–of–coolant accident.

The design of the emergency core cooling system (ECCS) has been simplified by reducing the number of valves and using passive one-way rupture disks to separate the HTS from the ECCS. This improves the reliability of system operation in the event of a loss of coolant accident (LOCA). These improvements also reduce the capital cost as well as significantly reducing the operating and maintenance costs for testing, inspection, maintenance and repair over the lifetime of the NPP.

Containment System

The CANDU 9 reactor building is a steel-lined, pre-stressed concrete structure which provides biological shielding and the containment boundary. A representative reactor building section view is shown in Figure 5-1.

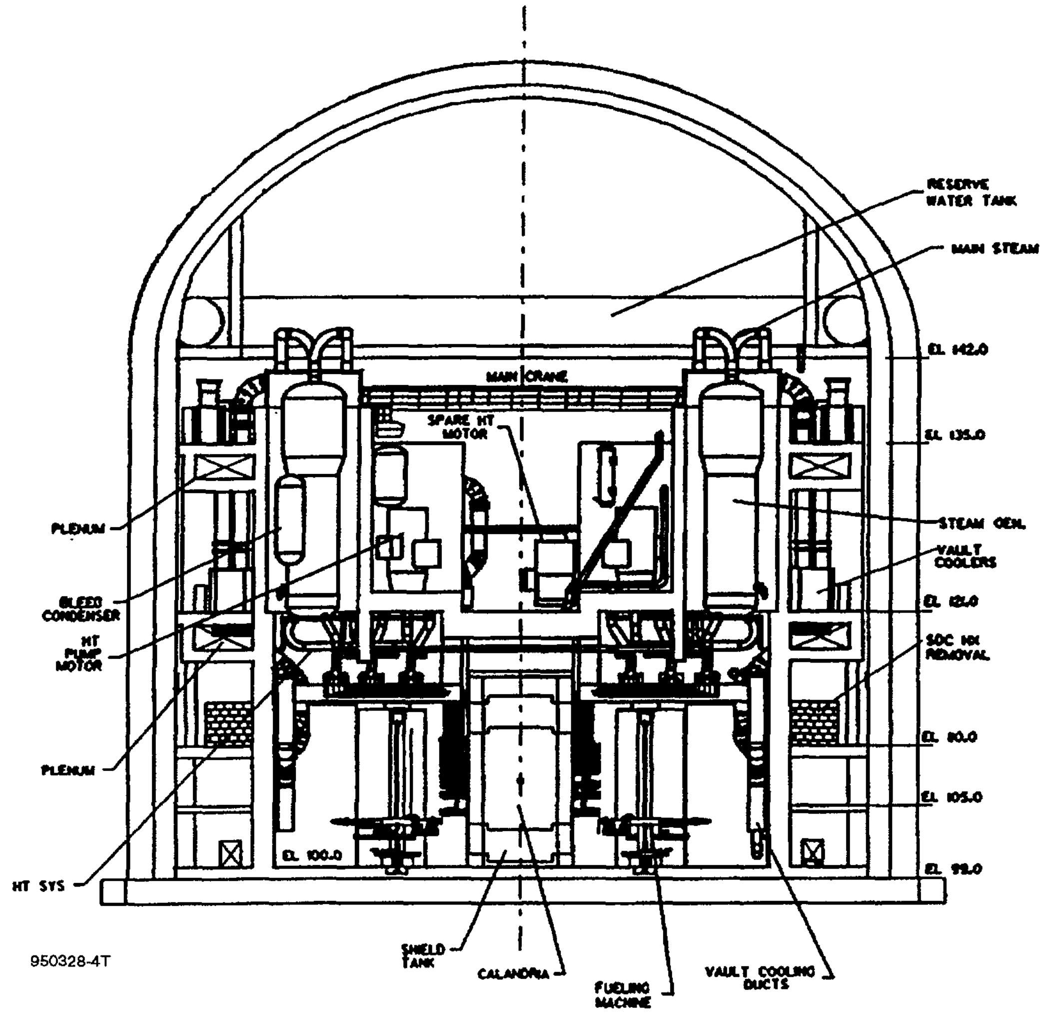

Figure 5-1. Reactor building sectional view.

The improved CANDU 9 containment system uses a 'large dry' cylindrical steel lined containment without a dousing system to achieve enhanced containment integrity with increased simplicity. The design leak rate is 0.2 % volume/day at design pressure. Because of the lower design leak rate from containment, the exclusion area radius for the siting of CANDU 9 can be as small as 500 meters, significantly reducing site area requirements for CANDU 9 plants. This is an important advantage in the context of siting requirements and land availability.

The free volume inside the containment is sufficiently large such that a pressure suppression system is not required in the short term to reduce the peak pressure after a LOCA below the design pressure. The long-term containment atmosphere heat sink is provided by the reactor building air coolers. Judicious layout of equipment inside containment results in large, open volumes, with good potential for natural circulation and no hydrogen traps.

The containment system automatically closes (i.e. buttons-up) all reactor building penetrations open to the containment atmosphere when an increase in containment pressure or radioactivity level is detected. Automatic isolation of the ventilation lines penetrating the containment structure has been enhanced and is provided by two separate and independent systems for increased reliability. The containment ventilation system provides enhanced atmospheric mixing with higher flow rates within the reactor building following a postulated loss-of-coolant-accident (LOCA). Passive catalytic recombiners are provided to control hydrogen concentration in the long term period after a LOCA; short-term control is accomplished by igniters.

5.3 Severe accidents (Beyond-design-basis accidents)

Probabilistic analysis has been a hallmark of CANDU safety philosophy since its inception; Canada remains one of the few countries whose regulations are based largely on probabilistic considerations.

For a loss of coolant accompanied by total failure of emergency core cooling system flow, the pressure tube will overheat, then sag or strain into contact with its surrounding calandria tube. Consequently, the fuel decay heat is transferred to the calandria tube through the pressure tube, and removed by the moderator. The surface finish of the calandria tube has been improved for CANDU 9 to enhance this heat transfer. A roughened surface is produced by shot-peening on the outside to promote nucleate boiling and a black oxide surface finish was added to the inside to increase radioactive heat transfer from the overheated fuel and pressure tube. Because of the cooling capability of the moderator, damaged fuel would remain within the pressure tubes, without UO_2 melting, so that the core geometry would be retained.

Should the moderator heat removal system subsequently fail, the CANDU 9 shield tank surrounding the calandria vessel provides an additional line of defence under severe accident situations. This will retain the debris inside the calandria, by keeping the outside of the calandria shell cool for a minimum period of 24 hours. This allows time for fission products to decay further, for decay heat to reduce, and for emergency planning. The role of the shield tank/end shield cooling system in severe accidents has been strengthened on CANDU 9 by provision of passive heat rejection to the reserve water tank.

Consistent with the targets for summed accident frequencies in the Electric Power Research Institute (EPRI) requirements document for Advanced Light Water Reactors, the following design requirements have been adopted for individual events during the CANDU 9 design process:

- A severe core damage accident shall be less frequent than 10^{-6}/year, and
- Any accident which causes a large radioactivity release shall be less frequent than 10^{-7}/year.

Compliance is confirmed by a comprehensive probabilistic safety assessment (PSA). High level accident management procedures are provided as part of the PSA, and accounted for in the event frequencies. The completed plant design is analysed to demonstrate that systems are capable of performing their assigned safety functions and that the radiation dose criteria are satisfied.

6. Plant buildings and layout

The conceptual plant arrangement, which includes the nuclear steam plant (NSP), and balance of plant (BOP), is designed as a stand alone, self-sufficient, single unit plant containing all facilities required for day-to-day operation. The basic building block consists of the reactor building (RB), the reactor auxiliary building (RAB), a maintenance building (MB), and the turbine building (TB).

Multiple unit CANDU 9 stations are achieved using the single unit design as a building block. The improved layout which features a narrow 110m wide "footprint" allows several units to be constructed adjacent to each other to form a very compact mult-unit station. A station services building (SSB) houses common services which are shared between units such as access control and security, change rooms, laundry services, central stores, overhaul facilities and large shops, heavy water management facilities, liquid waste management, engineering, maintenance and administrative services, and cafeteria facilities. The basic SSB is capable of serving a two unit station, and is easily expandable through the addition of floors and bays, to service additional units.

A typical two unit CANDU 9 station layout is shown in Figure 6-1.

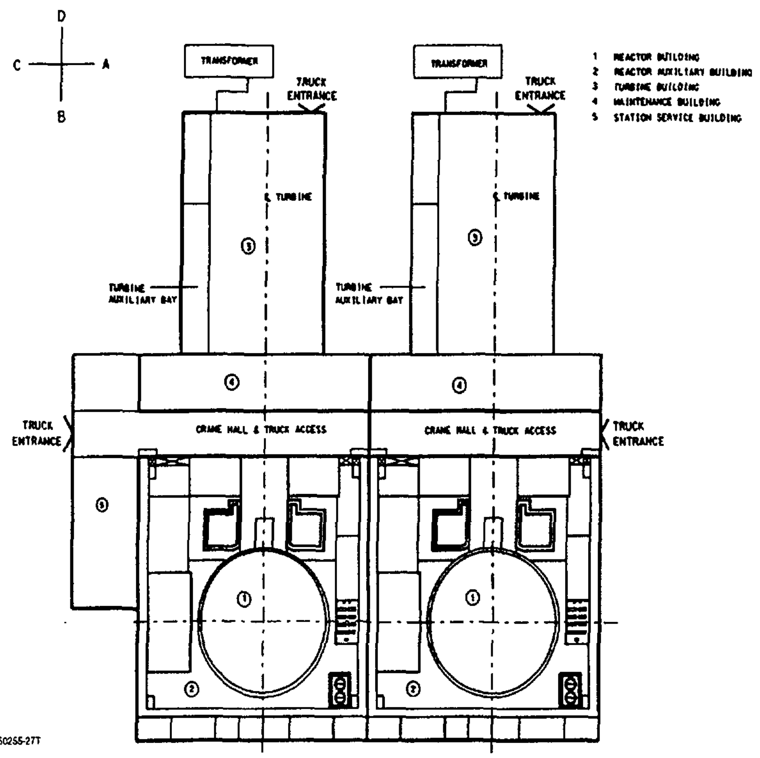

Figure 6-1. 2 unit CANDU 9 plant layout.

The reactor building comprises a cylindrical perimeter structure with a domed top and internal structures all on a common base slab. The reactor building houses the major nuclear steam plant systems and equipment. The major internal structures are reinforced concrete. The layout of the internal concrete structure walls, floors, active equipment and components are arranged to minimise personnel exposure to radiation while maximising access for testing and maintenance of components, and to minimise construction costs and schedule.

The reactor auxiliary building is a four storey structure consisting of a reinforced concrete substructure and a braced steel frame superstructure. The reactor auxiliary building is seismically qualified to design basis earthquake, and tornado protection is provided in accordance with the site requirements. Internally, it is divided into Group 1 and Group 2 areas.

The Group 1 area of the reactor auxiliary building accommodates umbilicals which run between the principal structures, the main control room, the irradiated fuel storage bays and associated fuel handling facilities, the irradiated fuel bays cooling and purification system, and the Group 1 recirculated cooling water system.

The Group 2 areas of the building house two of the special safety systems, safety support systems, the secondary control area, the Group 2 feedwater storage tank and pumps, the Group 2 electrical, control and monitoring systems, the Group 2 raw service water valve stations, and the ECC system recovery pumps. The secondary control area provides control and monitoring capability for all the systems required for the safe shutdown and monitoring of the plant and the maintenance of a long term heat sink should the main control room become uninhabitable or non-functional.

The turbine building is located on the 'D' side of the maintenance building with the turbine shaft alignment perpendicular to the reactor building thus assuring that any turbine generated missiles will not impact on containment or on the areas of the reactor auxiliary building which contains the main control room and secondary control area. This is also the optimum location with respect to ease of access to the control room, and the cost of piping and cable tray runs.

7. Technical data

General Plant Data		
Power plant output, gross	935	MWe
Power Plant output, net	875	MWe
Reactor thermal output	2 716	MW
Power plant efficiency, net	32.2%	%
Cooling Water temperature	25.5	°C

Nuclear Steam Supply System		
Number of coolant loops	1	
Primary circuit volume	220	m^3
Steam flow rate at nominal conditions	1 328.4	kg/s
Feedwater flow rate at nominal conditions	1 329.6	kg/s
Steam temperature	265	°C
Steam pressure (gauge)	5.0	MPa
Feedwater Temperature	177	°C

Reactor Coolant System		
Primary coolant flow rate	11 000	kg/s
Reactor outlet header operating pressure	9.9	MPa
Reactor inlet header operating temperature	267	°C
Reactor outlet header operating temperature	310	°C
Specific Enthalpy Change Across Core (average)	246	kJ/kg

Reactor Core		
Active core length	5.944	m
Core Radius, Effective	3.532	m
Number of Lattice Cells	480	
Cell array	square	
Lattice pitch	285.75	mm
Fuel inventory	110	t U
Fuel material	Sintered U_2O	
Fuel assembly(fuel bundle) total length	495.3	mm
Fuel design (rod array)	circular array	
Number of Fuel Bundles	5760	
Number of fuel rods (pins) /assembly	37-element	
Enrichment of fuel	0.7	Wt%
Average discharge burnup of fuel	8 520	MWd/t
Cladding tube material	zirconium alloy	
Cladding tube wall thickness	0.42	mm
Outer diameter of fuel rods	13.08	mm
Overall weight of fuel bundle	24.1	kg
Number of shutdown/control units - vertical	58	
Number of shutdown units - horizontal	8	
Reactivity control material - out of core	stainless steel	
- in core	zircaloy/stainless steel/cadmium	

Reactor Vessel		
Calandria shield tank assembly (CSTA) diameter	13.38	m
Wall thickness of cylindrical shell - calandria	32	mm
- shield tank	38.1	mm
CSTA overall length	8.2	m
Base material - Calandria	304L Stainless Steel	
Base material - shield tank	carbon steel	
Design pressure/temperature	1.2/105	MPa/°C
Transport weight (CSTA)	552	t

Fuel Channels		
Pressure tube inside diameter	103.4	mm
Number of pressure tubes	480	
Flow in maximum power channel	26	kg/s
Est. pressure drop across 12 bundles	830	kPa

Steam Generators		
Steam generator type	vertical U-tube with integral preheaters	
No. of steam generators	4	
Heat transfer surface each	5 138	m^2
Tube dimensions (outer dia./thickness)	15.9/1.13	mm

Maximum outer diameter	4 410	mm
Total height	20 822	mm
Transport weight	340.7	t
Shell and tube sheet material	carbon steel	
Tube material	Incoloy 800	

Reactor Coolant Pump

Number	4	
Design pressure (gauge)	12.7	MPa
Design temperature	279	°C
Design flow rate(at operating conditions)	3.2	m^3/s
Operating Temperature	267	°C
Pump Head	247.6	m
Power Demand at Coupling, hot	7 640	kWe
Pump casing material	carbon steel	
Pump speed	1 800	rpm

Pressurizer

Total volume	130	m^3
Steam volume: full / zero power	30/60	m^3
Design Pressure (gauge)	11	MPa
Design Temperature	316	°C
Heating power of heater rods	2.1	MWe
Inner diameter	3 000	mm
Total height	20 819	mm
Material	carbon steel	
Transport weight	187.7	t

Pressurizer Relief Tank

Bleed Condenser Tank - total volume	25	m^3
Design pressure/temperature	12.65/310	MPa/°C
Inner diameter (vessel)	2 496	mm
Total height	7 620	mm

Material (shell side)	carbon steel	

Primary Containment

Containment type	prestressed concrete with steel liner	
Overall form (spherical/cyl.)	cylindrical	
Inside Diameter	57	m
Wall thickness	1.5	m
Height (top of base slab to top of dome)	71.5	m
Free volume	124 000	m^3
Normal operating temperature range	15-50	°C
Design pressure (LOCA) (gauge)	210	kPa
Steam main break design condition (gauge)	450	kPa
Design leakage rate per day	0.2	vol%/day
Is secondary containment provided?	No	

Reactor Auxiliary Systems

Reactor water cleanup - (purification)	14.3	kg/s
Reactor water cleanup - filter type	disposable cartridge	
Residual heat removal - at high pressure	48	kg/s
Residual heat removal - at low pressure	580	kg/s
Coolant injection - at high pressure from accumulator	3 900	kg/s
Coolant injection - at low pressure	1 200	kg/s

Power Supply Systems

Main Output Transformer -rated voltage	345/24	kV
Main Output Transformer -rated capacity	1 050/1 150	MVA
Unit Service Transformer - rated voltage	24/13.8/13.8	kV
Unit Service Transformer - rated capacity	60/80/100	MVA

System Service Transformer -rated voltage	345/13.8/13.8	kV
System Service Transformer -rated capacity	60/80/100	MVA
Medium voltage busbars (13.8kV & 4.16kV))	10	
Number of low voltage busbar systems (480V)		
- Group 1 (non-safety)	3	
- Group 2 (safety)	2	
Standby diesel generating units		
- Group 1 (non-safety) number/rated power	2 / 9.5	MWe
- Group 2 (safety) - number/rated power	2 /4.5	MWe
Number of diesel-backed busbar systems	2	
Voltage level of these	4.16 / 0.48	kV ac
Number of battery-backed busbar systems	4	
Voltage level of these	480/208/120	V ac

Turbine Plant

Number of turbines per reactor	1	
Type of turbine	Tandem Compound	
Number of turbine sections per unit	HP/LP/LP/LP	
Turbine speed	1 800	rpm

Generator

Type	4 pole H_2 cooled	
Rated power	1 100	MVA
Active power	935	MWe
Voltage level of these	24	kV
Frequency	60	Hz

Condenser

Cooling water flow rate	46 000	L/s
Cooling water temperature	22	°C
Condenser pressure	47	mmHg

Condensate Pumps

Number	2 x 100%	
Flow rate	947.3	kg/s

Feedwater Tank

Volume	400	m^3
Pressure/temperature	0.75/154	MP/°C

Feedwater Pumps

Number of main pumps	3 x 50%	
Flow rate	665.8	kg/s
Feedwater temperature	177	°C
Number of auxiliary pumps	1 x 4%	
Flow rate	53	kg/s
Number of Group 2 auxiliary pumps	2 x 4%	

Condensate Feedwater Heaters

Number of heating stages	6	
Redundancies	2 parallel paths	

8. Project Status And Planned Schedule

The Basic Engineering Program (BEP) followed the CANDU 9 product design requirement definition work and conceptual studies. The BEP was a 39 month program started in January, 1995. The scope included performing up-front design engineering and carrying out a licensability review of CANDU 9 by the Atomic Energy Control Board (AECB).

During the basic engineering program, emphasis was put on the detailed design for the special safety systems and the associated key safety related systems. Sufficient engineering has been completed such that we can successfully meet the target schedule duration for construction with minimal technical and licensing risks. All space allocation has been completed for the plant buildings. Major systems have had their complete 3-D layouts incorporated into the plant CADD model. This work provided an up-front definition of improvements or enhanced features that are different from the reference plants, systems and components.

The CANDU 9 has undergone a thorough review by the AECB staff to obtain assurance that the CANDU 9 design is licensable in Canada. In 1997 January, the two year licensing review by the AECB was completed [6] and the results can be summarized by the following words from the AECB report: "AECB staff conclude that there are no fundamental barriers to CANDU 9 licensability in Canada."

With the Basic Engineering Program (BEP) objectives met in 1998 March, AECL continues to prepare for the delivery of the first CANDU 9 project. The current pre-project work includes further detailed engineering, safety analysis and equipment design engineering. In addition, this work will have the added benefit of keeping the team working with the advanced engineering tools and work processes so that we are ready to launch the first CANDU 9 project.

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GENERAL DESCRIPTION OF ADVANCED HEAVY WATER REACTOR

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Abstract

Advanced Heavy Water Reactor is a boiling light water cooled, heavy water moderated and vertical pressure tube type reactor with its design optimised for utilisation of thorium for power generation. The core consists of $(Th\text{-}U^{233})O_2$ and $(Th\text{-}Pu)O_2$ fuel with a discharge burn up of 20,000 MWd/Te. This reactor incorporates several features to simplify the design, which eliminate certain systems and components. AHWR design is also optimised for easy replaceability of coolant channels, facilitation of in-service inspection and maintenance and ease of erection. The AHWR design also incorporates several passive systems for performing safety-related functions in the event of an accident. In case of LOCA, emergency coolant is injected through 4 accumulators of 260 m^3 capacity directly into the core. Gravity driven water pool of capacity 6000 m^3 serves to cool the core for 3 days without operator's intervention. Core submergence, passive containment isolation and passive containment cooling are the added features in AHWR. The paper describes the various process systems, core and fuel design, primary components and safety concepts of AHWR. Plant layout and technical data are also presented. The conceptual design of the reactor has been completed, and the detailed design and development is scheduled for completion in the year 2002.

1. INTRODUCTION

The Advanced Heavy Water Reactor (AHWR) is a 235 MWe heavy water moderated, boiling light water cooled, vertical pressure tube type reactor with its design optimised for utilisation of thorium for power generation. The conceptual design and the design feasibility studies for this reactor have been completed and at present the reactor is in the detailed design stage. The reactor design has a number of passive features described subsequently in this paper. The overall design philosophy includes achievement of simplification to the maximum extent.

While the detailed economics of operation of AHWR are yet to be worked out, pending finalisation of plant design, the reactor incorporates several features to simplify the design and to eliminate certain systems and components, likely to make AHWR economically competitive with other available options for power generation. Some important elements in the AHWR design, having bearing on its improved economics, are as follows:

- Elimination of high pressure heavy water coolant thereby leading to reductions in heavy water inventory, heavy water leaks, and exposure of personnel to tritium.
- Replacement of complex and long delivery items like steam generator by steam drum of simple construction.
- Minimising dependence on active systems like primary coolant pumps (due to natural circulation of light water coolant), thus enabling usage of conventional equipment for performing duties of much less safety importance.
- Shop fabrication of major components of the reactor, such as coolant channels, to reduce construction cost and time.

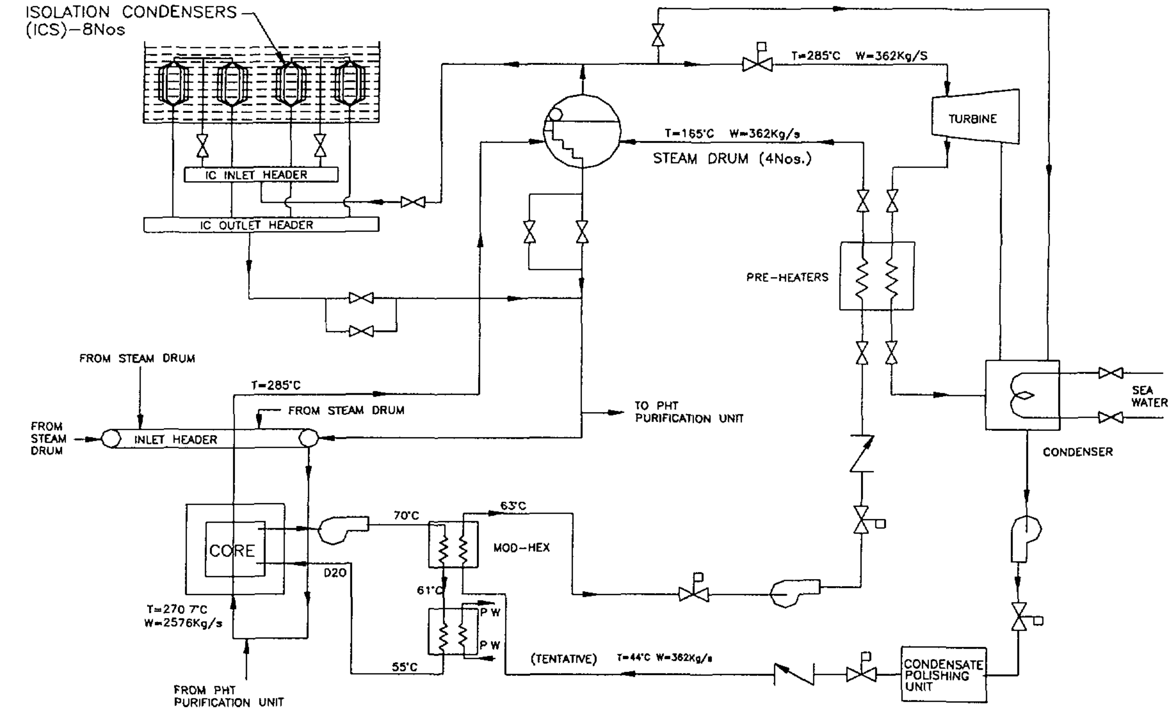

FIG. 1. Simplified PHT system flow sheet.

2. DESCRIPTION OF THE NUCLEAR SYSTEMS

2.1 Primary Heat Transport System

The Primary Heat Transport (PHT) System is shown in Fig. 1. This system is designed to cool fuel assemblies by boiling light water, which flows through the coolant channels by natural circulation.

The steam-water mixture from each coolant channel is led through 125 mm NB tail pipes to four steam drums, which are located with an elevation difference of 39 m with respect to inlet feeder (coolant channel bottom). The steam at a pressure of 70 kg/cm^2 is separated from steam-water mixture in steam drums. The steam is led to the turbine by two 400 mm NB pipes. The steam from the turbine is condensed and after purification of the condensate and preheating, it is pumped back to steam drums at a temperature of 165°C. The feedwater is mixed with the water separated from steam-water mixture at 285°C in steam drums. The water level in the steam drum is a function of reactor power and is maintained at a set level during power operation.

A nearly uniform exit quality of steam in all the channels is maintained by providing orifices at the bottom of the reactor coolant channels.

2.2 Core Decay Heat Removal System

The reactor core decay heat removal through Isolation Condensers (ICs) is a passive safety feature for the removal of the core decay heat during normal reactor shut down. The system is designed to remove the core decay heat for a period of three days without operators' intervention.

The core decay heat removal system is designed for the removal of heat at 3% of full reactor power, with steam temperature of 150°C, and has capability for the removal of decay heat of 6% of full power for a duration of a few seconds, when the steam temperature is 285°C. The IC consists of vertical tubes, joined at both ends to cylindrical headers and is submerged in the Gravity Driven Water Pool (GDWP). The steam from coolant channels enters IC tubes from top end through steam drums and is condensed due to surrounding cool water of GDWP. The condensate returns by gravity to the PHT system through a outlet header of ICs.

The system is designed for 4 x 50% capacity. Eight ICs (out of which four operate at a time) are located in eight compartments of GDWP. The capacity of GDWP (for cooling) is based on the requirements of 2 m of water level above the ICs.

2.3 Active Shut Down Cooling System

An active shutdown cooling system is provided to lower the temperature of PHT system from 150°C to 60°C during long shut down of the reactor for maintenance. The system consists of four loops out of which two operate at a time. This system is designed to take care of the eventuality of non-availability of ICs for removal of the reactor core decay heat.

2.4 Moderator System

The moderator system is designed as a full tank concept for normal operation. However, the level control system is incorporated to change heavy water level in the top reflector region for reactor power adjustments. Helium is used as a cover gas in AHWR.

2.5 Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) is designed to remove the core heat by passive means in case of a postulated Loss of Coolant Accident (LOCA). In the event of rupture/breakage in the primary coolant pressure boundary, the cooling is achieved initially by a large flow of borated light water from advanced accumulators and later cooling of the core is achieved through the GDWP. The inventory of GDWP is adequate to cool the reactor core for a period of three days without operators' intervention.

The ECCS consists of four accumulators of total capacity of 260 m^3 and a gravity driven water pool of capacity 6000 m^3, both connected to the ECC header. The ECC header is connected to individual coolant channels above the tail pipe. The ECC coolant enters the core through eight perforated water tubes arranged in the fuel cluster so as to ensure wetting of fuel pins by spray action. The coolant, after coming out of rupture pipe gets accumulated in the reactor cavity along with PHT coolant and is re-circulated through heat exchangers to ensure long term core cooling.

2.6 Reactor Core and Fuel Design

2.6.1 Design Objectives

The reactor physics parameters are finalised to meet the following important design objectives:

- Power in Thorium fuel: 75% , approximately.
- Slightly negative void coefficient of reactivity.
- Discharge burn up: Minimum target of 20,000 MWd/Te
- Initial plutonium inventory: As low as possible.
- Self sustaining in U^{233}.
- Thermal power: 750 MW.

2.6.2 Fuel Cluster Design

The reactor core has 424 coolant channels. In 340 channels, the fuel cluster consists of 44 (Th-U^{233})O_2 and 8 (Th-Pu)O_2 pins, called Thoria and MOX fuel pins, respectively. The fuel clusters in the remaining 84 channels have all Thoria pins. To generate lower power fraction in MOX fuel, the plutonium content in MOX is kept low at 4.5 % (typically) and these pins are located at the outermost circle of the fuel cluster. Since the MOX fuel accumulates burn up faster than the Thoria pins, they need replacement from reactivity considerations, so as to allow Thoria pins to reach the designed burn up of 20,000 MWd/Te. The fuel reconstitution frequency is estimated to be two times in its life. The power flattening is achieved by loading fresh MOX fuel clusters in the outer most radial zone and the reconstituted MOX fuel clusters in middle & inner zones. The maximum. channel power is 2.3 MW.

2.6.3 Moderator and Reflector

The reactor core is contained in a calandria having heterogeneous mixture of heavy water as moderator and pyrocarbon material as scatterer. Heavy water is provided as reflector in the radial direction with thickness of 300 mm. Heavy water is also reflector in the axial direction with thickness of 750 mm at bottom location and 600 mm thick at top location. This arrangement, evolved after detailed analysis of a number of cases, meets the requirements of satisfactory k-effective value and negative void coefficient of reactivity.

2.6.4 2.6.4 Shut Down System

AHWR is provided with two independent fast acting shut down systems namely primary and secondary shut down systems. The primary shut down system (PSS) consists of absorber rods having boron carbide as neutron absorbing material. Boron carbide is filled in an annulus of thickness 1.5 mm, formed by stainless steel shells. The secondary shut down system consists of a liquid poison injection system in which lithium pentaborate solution with boron content of 20 g/litre will be injected in the poison tubes.

2.6.5 Reactor Fuel Design

2.6.5.1 Design Objectives

The fuel assembly of AHWR is designed to provide :

- Continuous full power operation.
- Low pressure drop of the coolant.
- Stable neutronic/thermal hydraulic coupling during all stages of reactor operations.
- On-power fuelling operation.
- Reconstitution of fuel clusters
- Spray on fuel pins from ECC system during LOCA.

2.6.5.2 Description

The fuel assembly consists of components like fuel cluster and shield plug. The fuel cluster of length 4200 mm is suspended inside the pressure tube of coolant channel from top by a hanger assembly and has features to enable the separation of the shield plug from the spent fuel inside the fuelling machine and also joining of the same shield plug with new fuel.

The fuel pins are arranged in a square lattice pitch of 13.7 mm in the cluster. The fuel pin of 11.2 mm OD consists of Zircaloy clad tube of thickness 0.6 mm. In addition to fuel pins, the fuel cluster has eight perforated Zircaloy tubes (called water tubes) which are arranged at outer periphery, for spraying emergency core cooling water directly on the fuel pins, during LOCA. The fuel pins are held between top and bottom tie plates with the help of eight fuel pins, called tie rods. The remaining fuel pins (44 nos.) are resting on the bottom tie plate and are free to expand axially at top. The inter-element spacing of 2.5 mm between fuel pins is maintained with the help of six Zircaloy spacers.

2.7 Fuel Handling and Transport System

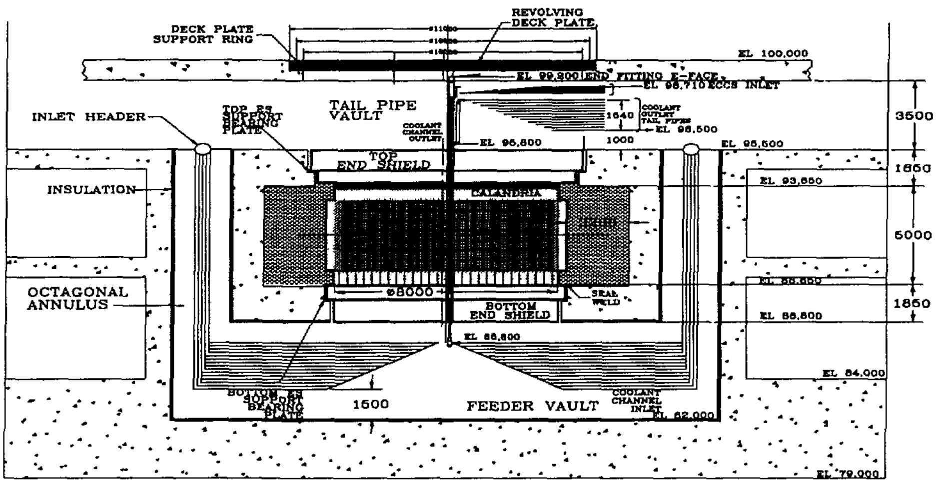

FIG. 2. AHWR reactor component layout.

2.7.1 Design Philosophy

The AHWR is designed to have on-power fuel handling feature to increase the capacity factor of the reactor by maintaining the designed reactivity in the core and optimising the fuel burn up. The fuelling frequency is estimated to be six assemblies in a month.

The fuelling machine has the following major components:

- A fuelling machine head for handling the fuel clusters by means of ram drives and snout drive for coupling and making pressure tight joint with the coolant channel.
- A carriage for movement of the machine on rails, laid between reactor block and storage block.
- Shielding to limit the surface dose rate below 0.6 mr/hr, by using lead, steel and paraffin wax as shielding materials, so as to make the machine and reactor top approachable during fuelling operations.
- A cooling system to remove the decay heat from the fuel clusters.
- A control and an electrical system for remote operation of the machine from the control room in auto and manual mode.

2.8 Primary Components

The primary components of reactor pile block structure consist of equipment and components contained in the reactor cavity. These include calandria with vertical coolant channels, end shields at top and bottom, concrete vault filled with light water, tail pipe vault at top and feeder vault at bottom. Figure 2 shows the layout of AHWR reactor components

2.8.1 Design Bases

The following are the major bases for the design of AHWR pile block components and structures:

- Easy replaceability of coolant channels.
- Heterogeneous moderator and reflector system having heavy water and pyrocarbon material.
- Features to facilitate in -service inspection and maintenance.
- Ease of erection.

- Adequate shielding to enable accessibility to areas outside the pile block during reactor operation.
- Provision for direct emergency core cooling.
- On-power refuelling.

2.8.2 Calandria

The calandria is a vertical cylindrical shell structure having a sub-shell at each end, connected by flexible annular plate. Both the sub shells are in-situ welded to tube sheets of end shields. Vertical calandria tubes are arranged in a square lattice pitch and rolled to lattice tubes of end shields. Nozzles and penetrations are provided in the shell and sub-shell regions of calandria for circulation and level control of heavy water moderator and circulation of helium (cover gas). Vertical penetrations are provided for primary and secondary shut down systems, and reactivity mechanisms. The calandria is provided with over-pressure relief devices to mitigate pressure rise in an accident.

2.8.3 Coolant Channel

The coolant channel accommodates fuel assembly, maintains thermal insulation between hot pressure tube and cold calandria tube and provides interface for coupling to primary heat transport system at both ends. It also facilitates injection of light water directly into the fuel clusters from emergency core cooling system in case of LOCA and provides interface to facilitate on-power fuelling operations.

The design provisions are made to take care of:

- Thermal expansions.
- Creep/growth related dimensional changes.

The coolant channel consists of a pressure tube, end fittings at top and bottom ends. The coolant channels are supported on top end shield. The top end fitting has a provision for connection to outlet tail pipe and ECCS injection pipe. It also has suitable features to enable engagement of the fuelling machine. The bottom end fitting is connected to the inlet header, which is located above the core, through individual feeder pipe. Calandria tubes, concentrically located outside the pressure tubes in the calandria region, are rolled to lattice tubes of end shields.

2.8.4 End Shields

End shields are provided at top and bottom ends of the calandria and are in-situ welded to calandria sub-shells. The end shields are designed to achieve a dose rate less than 0.6 mr/hr, in the tail pipe vault and the feeder vault, after one hour of reactor shut down, to allow access of personnel in these areas. The shielding materials are arranged in different layers like steel, water and mixture of water & carbon steel balls. The top end shield supports coolant channels and other vertical penetrations. The top end shield is provided with a composite tube sheet at the bottom end for circulation of heavy water, to remove the heat generated in the composite tube sheet of end shield. Both the end shields have recirculation-cooling system using light water so as to maintain a temperature of 55°C from thermal stress considerations. The end shields are supported through bearing plates on the concrete structure of the reactor block.

2.8.5 Deck Plate

The deck plate provides shielding above tail pipe vault to limit the dose rate below 0.6 m/hr, during full power operation, so as to make the reactor top accessible for on-power fuelling and other operations. The deck plate serves as a platform for removal of fuel assemblies and supports the shielding skirt of the fuelling machine. The deck plate consists of inner and outer revolving floors that are supported on special bearings to facilitate alignment to any lattice position by selecting a proper combination of rotation of revolving floors. The inner revolving floor has a central opening of 600 mm diameter which is normally closed by a shielding plug and a flapper mechanism. During fuelling operations, the shielding plug is removed and the flapper is opened after lowering of shielding skirt of the fuelling machine. The shielding skirt also makes a leak tight joint with the inner revolving floor.

2.8.6 Reactor Vault

The calandria is surrounded by a heavy density concrete vault, filled with light water, to provide thermal and biological shielding against neutrons and gamma rays. The thickness of water shield and concrete are arrived on the basis that the dose rate in adjacent rooms is less than 0.6 m/hr during reactor operation and in the annulus after one hour of reactor shut down. The vault cooling system is designed to remove the heat generated in vault water due to attenuation of gamma rays and due to transfer of heat from calandria. The inlet and outlet piping of calandria vault are provided with inverted 'Us' to prevent draining of the vault in case of a pipe break/rupture.

3. DESCRIPTION OF TURBINE GENERATOR PLANT SYSTEM

3.1 Steam and Feedwater System

3.1.1 Design Requirements

The steam and feedwater system is a closed loop system designed to meet the following design requirements:

- Generation of 99.9% dry steam in steam drums for operation of the turbine.
- Condensation of steam in the condenser which is exhausted from the turbine in operation mode or in by-pass mode.
- Purification of full flow of condensate and pump back to steam drums through pre-heaters and feed water pumps, which are conventionally available equipment.
- Acts as a heat sink for the reactor under emergency conditions.

3.1.2 Steam Drum and Steam System

The system has four steam drums, constructed from carbon steel and lined with stainless steel of overall size 3600 mm diameter x 10,000 mm length. Each steam drum is connected to 106 tail pipes coming from the reactor coolant channels, which carry steam-water mixture. The water level in each steam drum is controlled by a water level regulator by comparison with set point level and flow of feed water with respect to steam flow rate.

The steam from each steam drum is tapped from top location by a 300 mm NB pipe. The outlet pipes from two steam drums are connected to a 400 mm NB pipe and two of these pipes (from four steam drums) are connected to steam chest of the turbine. The pressure relief system (consisting of four safety valves and four relief valves) is installed on 400 mm NB pipe lines within the primary containment of the reactor to protect against over pressure in case of rupture of the pipe line.

4. INSTRUMENTATION & CONTROL SYSTEMS

4.1 Design Concepts

The function of the instrumentation and control system is to monitor and control various plant parameters like neutronic, thermal hydraulic, and process parameters reliably, using principles of redundancy, diversity, testability and maintainability. The above is achieved by having triplicated channels, using principle of 2 out of 3 logic and fail safe criteria for the safety systems. The system is also provided with a feature of on-power testing of channels. The instrumentation for control and protection system is independent and separate. An extensive operator information system is provided to have features like display, alarm, record, retrieval of the plant parameters etc. The details of this system are being worked out.

4.2 Reactor Protection System

The shutdown system designed for fast transients consists of two completely independent and redundant devices. The fast acting primary shut down system consists of mechanical shut off rods and secondary shut down system to inject liquid poison into the tubes.

4.3 Reactivity Control

The reactivity is controlled by the following methods:

- Refuelling to take care of reactivity loss due to fuel depletion.
- Addition of poison into the moderator to control long term excess positive reactivity by addition of boron.
- Moderator level control in top reflector region to effect power changes.
- Adjuster rods for Xenon override operations.

5. ELECTRICAL SYSTEMS

The salient features of the electrical system are as follows.

- Minimum two independent off-site power sources of 220 kV for start-up through one Start-up Transformer (SUT).
- Two independent power supply sources for normal power operation
  - From grid through Start-up Transformer (SUT).
  - From main generator through Unit Transformer (UT).
- Automatic transfer of station auxiliaries to other source in case of failure of one source.
- Three Class 1E emergency diesel generators, one feeding to each of the two independent bus section and one stand-by to either of the bus section to provide on-site stand-by power for Class 1E equipment.
- Three independent Class 1E 240V AC system with a stand-by and automatic switching and battery back up for reactor protection channel.
- Three independent 2 x 100%, 48 V DC system with battery back-up for reactor protection channel.
- AC voltage levels of 6.6 kV and 415 V.
- DC voltage levels of 220 V and 48 V.

6. SAFETY CONCEPTS

6.1 Safety Requirements and Design Philosophy

Prevention of accidents is the basic design philosophy in AHWR. All proven measures of current safety concepts assuring reliable operation are incorporated in the design to prevent accidents. These include the following:

- Systems and components designed with conservative margins.
- Use of the redundancy concept for operating systems to increase the reliability.

- Preventive maintenance.
- In-service inspection.
- Large water reservoir in GDWP.
- Incorporation of passive safety features.
- Negative void coefficient of reactivity.

6.2 Safety Systems

6.2.1 Passive Safety Features

The Advanced Heavy Water Reactor (AHWR) is being designed with incorporation of many passive systems/elements to facilitate the fulfilment of safety functions e.g. reactor operation, reactor shutdown, residual heat removal, emergency core cooling, confinement of radioactivity etc. For removal of heat from the reactor core under operating as well as accident conditions, heat removal paths and systems are shown in Figure 3. These systems are described in the following paragraphs.

6.2.2 Natural Circulation of Primary Coolant

During normal reactor operation, full reactor power is removed by natural circulation caused by thermosyphoning phenomenon. Primary circulation pumps are eliminated and the necessary flow rate is achieved by locating the steam drums at a suitable height above the centre of the core, taking the advantage of reactor building height. By eliminating nuclear grade primary circulating pumps, their prime movers, associated valves, instrumentation, power supply and control system, the plant is made simpler, less expensive, easier to maintain as compared to options involving forced circulation in the primary coolant circuit. The above factors also lead to considerable enhancement of system safety and reliability since pump related transients have been removed.

6.2.3 Core Decay Heat Removal

During normal reactor shut down core decay heat is removed by Isolation Condensers (ICs) which are submerged in Gravity Driven Water Pool (GDWP), located above the steam drum. The steam, led to the ICs by means of natural circulation, condenses inside the IC pipes and heats up the

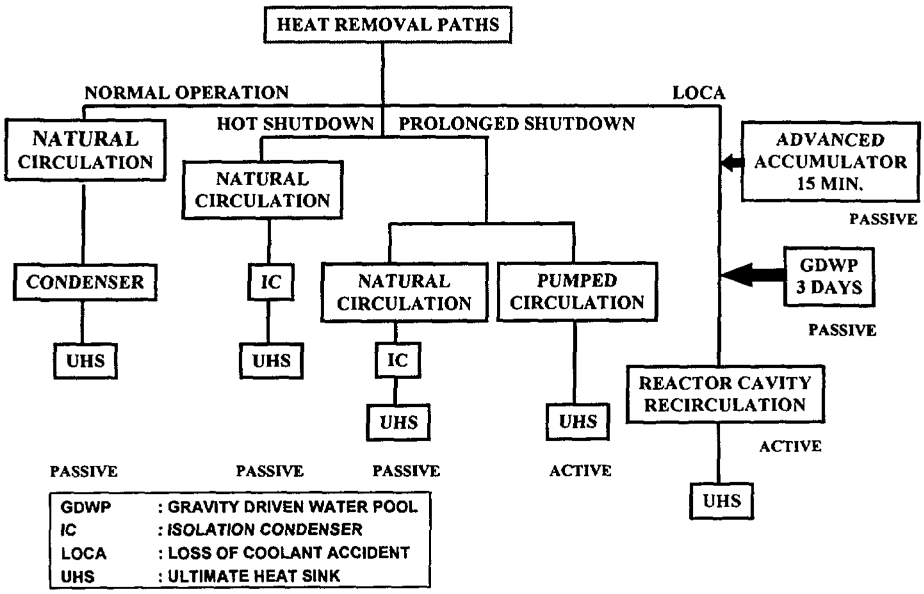

FIG. 3. Heat removal paths of AHWR..

surrounding pool water. The condensate returns by gravity to the core. The water inventory in GDWP is adequate to cool the core for more than 3 days without any operator intervention and without boiling of GDWP water. A GDWP cooling system is also provided. An Active Shut Down Cooling System (ASDCS) is also provided to remove the core decay heat in case the ICs are not available.

6.2.4 Shutdown Systems

Two completely independent and redundant fast acting devices (mechanical shut off rods and liquid poison injection) are provided to shutdown the reactor. These devices are actuated by active systems. In case of failure of these devices to act, the reactor will be shutdown due to negative void coefficient of reactivity.

6.2.5 Emergency Core Cooling

During Loss-Of-Coolant-Accident (LOCA) emergency coolant injection is provided by passive means to keep the core flooded so as to prevent overheating of the fuel.

The Emergency Core Cooling System (ECCS) is designed to fulfil the following two objectives:

1. To provide large amount of cold borated water directly into the core in the early stage of Loss-of-Coolant Accident (LOCA) and then a relatively small amount of cold borated water for a longer time to quench the core. This objective is achieved through ECCS accumulator.
2. To provide water through Gravity Driven Water Pool (GDWP) to cool the core for more than 3 days.

Long-term core cooling is achieved by active means by pumping water from reactor cavity to the core through heat exchangers.

6.2.6 Core Submergence

After Loss Of Coolant Accident (LOCA), the water from the PHT system, advanced accumulators and the GDWP, after cooling the core, will be guided and get collected in the space around the core called reactor cavity. Thus the core will be submerged under water. If GDWP fails during any postulated scenario, its inventory will get collected in the reactor cavity and will provide heat sink to the core.

6.2.7 Failure of ECCS during LOCA

The reactor core of AHWR contains huge inventory of heavy water moderator and surrounding vault water. Although the possibility of failure of ECCS is very rare but if ECCS is not available during LOCA due to any reason, the fuel temperature will start rising and ballooning of pressure tubes will occur. Due to ballooning the pressure tubes will come in contact with calandria tubes and heat will be transferred to the moderator and then from moderator to vault water.

6.2.8 Passive Containment Isolation

To protect the population at large from exposure to radioactivity, the containment must be isolated following an accident. To achieve this, passive containment isolation, in addition to the closing of the normal inlet and outlet ventilation dampers, has been provided in AHWR. The reactor building air supply and exhaust ducts are shaped in the form of U bends of sufficient height. In the event of LOCA, the containment gets pressurised. This pressure acts on GDWP inventory and pours water by swift establishment of a siphon, into the ventilation duct U bends. Water in U bends acts as seal between the containment and the external environment, providing the necessary isolation between the two. Drain connections provided to the U bends permit the re-establishment of containment ventilation manually when desired.

6.2.9 Passive Containment Cooling

Passive containment coolers (PCCs) are utilised to achieve post-accident primary containment cooling in a passive manner, and to limit the post-accident primary containment pressure. A set of PCCs are located below the GDWP and are connected to the GDWP inventory. During LOCA, the mixture of hot air and steam is directed to flow over the PCCs. Steam condenses and hot air cools down at the PCC tube surface and hence provides long term containment cooling after the accident.

6.3 Severe Accidents

The primary objective followed in the development of AHWR is to enhance the level of safety to such an extent that the probability of occurrence of a severe accident becomes negligibly low, on account of the presence of safety features as already described. This will be confirmed by a PSA. In this context, it may be noted that the core submergence, discussed earlier, and presence of a large pool of water below the reactor, following an accident, will serve as effective barriers to escalation of any severe accident scenario.

7. PLANT LAYOUT

7.1 General Arrangement of the Reactor Building

The reactor building of AHWR is a cylindrical concrete structure. This consists of two coaxial cylindrical shells closed at top by dome structures. The inner structure, called primary containment, accommodates high enthalpy systems like reactor core, primary coolant systems, fuelling machine etc. The primary containment has a diameter of 44 m and a height of 72 m, and it is constructed from pre-stressed concrete. A large pool of water, called the Gravity Driven Water Pool (GDWP), is located near the top of the primary containment. This pool is designed to perform several passive safety functions. The outer structure having a diameter of 58 m and a height of 74.5 m is constructed with reinforced concrete and is called secondary containment. Both these structures are supported on a concrete raft. AHWR reactor building elevation is shown in Figure 4.

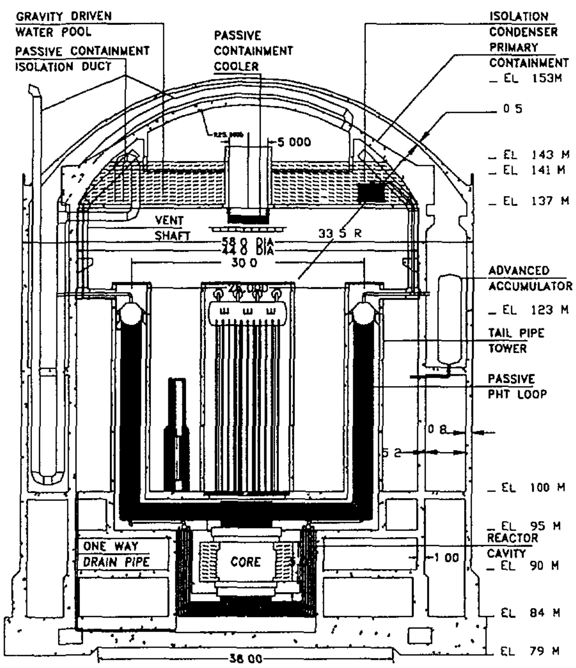

FIG. 4. AHWR reactor building elevation.

7.2 Criterion for Design of Layout

The layout of reactor building is carried out to meet the following objectives:

- Minimisation of primary containment volume.
- Effective utilisation of space in the annulus between primary and secondary containments.
- Unrestricted entry to reactor top for on- power fuel handling and transfer operations.
- Adequate shielding against radiation and preventing spread of radioactive contamination during normal and accidental conditions.
- Provision of a large water inventory at a suitable height, capable of supporting a number of passive systems.
- Submergence of reactor core under water before exhaustion of inventory of emergency core cooling system.
- Easy access to maximum number of equipment for operation and maintenance during normal and accidental situations.
- Fire prevention and control.

8. TECHNICAL DATA (As per format given in IAEA-TECDOC-968)

<u>General Plant Data</u>		
Power plant output, gross	245	MWe
Power plant output, net	235	MWe
Reactor thermal output	750	MWt
Power plant efficiency	33	%
Cooling water temperature	30	°C
<u>Nuclear Steam Supply System</u>		
Number of coolant loops	4	
Primary circuit volume	307	m^3
Steam flow rate at nominal conditions	362	kg/s
Feedwater flow rate at nominal conditions	362	kg/s
<u>Reactor coolant system</u> (Calculation being updated)		
Primary coolant flow rate	2576	kg/s
Reactor operating pressure	7	MPa
Steam temperature/pressure	285/7	°C/MPa
Feedwater temperature	165	°C
Core coolant inlet temperature	270.7	°C
Core coolant outlet temperature	285	°C
Mean temperature rise across core	14.3	°C
<u>Reactor Core</u>		
Active core height	3.5	m
Equivalent core diameter	7.6	m
Heat transfer surface in the core	2740	m^2
Total fuel inventory	49	t HM#
Average linear heat rate	9.6	kW/m
Average fuel power density	15.3	kW/kg HM
Average core power density (volumetric)	84	kW/l
Thermal heat flux, F_q (average)	274	kW/m^2
Enthalpy rise, F_H (average)	291	kJ/kg

Fuel material		$(Th\text{-}Pu)O_2 + (Th\text{-}U^{233})O_2$	
Fuel assembly total length		4027	mm
Rod array			
Number of fuel assemblies		424	
Number of fuel rods/assembly		52	
Number of control rod guide tubes		36	
Number of spacers		7	
Enrichment (range) of first core, average			Wt%
Enrichment of reload fuel at equilibrium core			
Plutonium		4.5 Wt%	
Uranium-233		Self sustaining	
Operating cycle length (fuel cycle length)			months
Average discharge burnup of fuel		20 000	MWd/t HM
Cladding tube material		Zircaloy	
Cladding tube wall thickness		0.6	mm
Outer diameter of fuel rods		11.2	mm
Fuel channel/box; material		Zr-2.5 wt% Nb	
Overall weight of assembly, including box		200	kg
Heavy metal weight/assembly		114.6	kg
Active length of fuel rods		3500	mm
Burnable absorber, strategy/material			
Number of control rods (assemblies) (RRC)		36	
Number of grey control rods (assemblies) (GRC)			
Number of water displacer rods (assemblies) (WDR)			
Absorber rods per control assembly			
Absorber material :	RRC		
	GRC		
	WDR		
Drive mechanism	RRC		
	GRC		
	WDR		
Positioning rate [or steps/s]			mm/s
Soluble neutron absorber			

\# HM = Heavy Metal

Reactor pressure vessel

Calandria

Inner diameter	8600	mm
Wall thickness	≈ 50	mm
Total height, inside	5000	mm
Base material	SS 304L	
Transport weight (empty)	55	t

Coolant channel

Total height of coolant channel	13400--15700	mm
Inner diameter of pressure tube	120	mm
Wall thickness of pressure tube	4	mm
Base material	Zr-2.5 wt% Nb	
Design pressure/temperature	8.5/300	MPa/°C
Transport weight (without fuel and water)	0.5	t

Reactor recirculation pump

(AHWR is Natural circulation system)	Not Applicable	
Type	-	
Number	-	
Design pressure/temperature	-	MPa/°C
Design mass flow rate (at operating conditions)	-	kg/s
Pump head	-	MPa
Rated power of pump motor (nominal flow rate)	-	kW
Pump casing material	-	
Pump speed (at rated conditions)	-	rpm
Pump inertia	-	kg m^2

Primary Containment

Type	Pressure suppression	
Overall form (spherical/cylindrical)	Cylindrical	
Dimensions (diameter/height)	44/72	m
Free air volume V_1 volume	9724	m^3
V_2 volume	53240	m^3
Design pressure/temperature	359/156	kPa/°C
Design leakage rate		vol%/day
Is secondary containment provided?	Yes	

Reactor auxiliary systems

Reactor water cleanup,	capacity	33	kg/s
	filter type	Mixed bed	
Residual heat removal,	at high pressure	22.5	MW
	at low pressure (150°C)	22.5	MW
Coolant injection,	at high pressure	1322	kg/s
	at low pressure	49	kg/s

Power supply systems

Main transformer,	rated voltage		kV
	rated capacity		MVA
Plant transformers,	rated voltage		kV
	rated capacity		MVA
Start-up transformers,	rated voltage		kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)			
Number of low voltage busbar systems			
Standby diesel generating units :	number		
	rated power		
Number of diesel-backed busbar systems			
Voltage level of these			V AC
Number of DC distributions			
Voltage level of these			V DC
Number of battery-backed busbar systems			
Voltage level of these			V AC

Turbine Plant

Number of turbines per reactor	1	
Type of turbine(s)	Horizontal, impulse reaction	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/1 LP	
Turbine speed	3000	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	6.8/284	MPa/°C

Generator

Type	Static exciter, stator and rotor core hydrogen cooled and stator windings water cooled.	
Rated power	275	MVA
Active power	245	MWe
Voltage	16.5	kV
Frequency	50	Hz
Total generator mass, including exciter	240	t
Overall length of generator	11	m

Condenser

Type	Surface condenser	
Number of tubes	21193 (18 BWG) 413 (16 BWG)	
Heat transfer area	21 020	m^2
Cooling water flow rate	17	m^3/s
Cooling water temperature	30	°C
Condenser pressure	8.4	kPa

Condensate pumps

Number	2	operating
	1	stand by
Flow rate	212	kg/s
Pump head		MPa
Temperature	42.8	°C
Pump speed	1500	rpm

Condensate clean-up system

Full flow/part flow	1440	m^3/h
Ion Exchanger type	Mixed bed	

Feedwater pumps

Number	2	operating
	1	stand by
Flow rate	260	kg/s
Pump head		MPa
Feed pump power		MW
Feedwater temperature (final)	165	°C
Pump speed	3000	rpm

Condensate and feedwater heaters

Number of heating stages,	low pressure	1
	high pressure	2

9. PROJECT STATUS AND PLANNED SCHEDULE

The conceptual design of AHWR was completed in December, 1997. On the basis of first level analytical studies and experimental work, the feasibility of the design concept was established and a Feasibility Report was issued.

Detailed design of nuclear systems of AHWR is in progress. It is planned to develop design details for nuclear systems, conduct supportive analysis and experimental developments, prepare detailed specifications for non-nuclear systems and issue Detailed Project Report in the year 2002.

IAEA-SM-353/58

REACTOR PLANT V-407

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Abstract

The (WWER-640) V-407 reactor plant represents a new generation reactor design for a Nuclear Power Plant (NPP) of medium power level, with a high level of reliability and efficiency, in comparison with the preceding standard series of plants with domestically designed reactors. The V-407 is a light-water cooled and moderated reactor intended for the generation and supply of steam to turbo-generator plants of nuclear installations of average power, for production of electricity with a frequency of 50 Hz and 640 MW (electric). It is intended for operation under base-load and load-follow conditions with peak and semi-peak modes, for location in regions with high seismic activity (up to magnitude 10). The safety concepts of the NPP with V-407 reactors conform to the worldwide trend in development of future generation nuclear power plant designs. According to the classification by the International Safety Advisory Group (INSAG) in the documents, INSAG-3 and INSAG-5, the V-407 reactor plant is in the class of evolutionary projects of medium-size NPP with passive safety systems.

1. INTRODUCTION

The development of the NPP with a V-407 reactor draws on extensive operating experience with systems and equipment of domestic and foreign NPPs with WWERs and PWRs. This work aims to achieve an optimal combination of proven systems and equipment and the application of new safety solutions, such as:

- double containment;
- passive systems for emergency core cooling;
- improvement of the inherent safety properties of the core; and
- improvement in reliability of safety barriers, including up-to-date methods of diagnostics.

Such a combination has made it possible to design a V-407 reactor plant with high safety indices and an acceptable level of economic indices, corresponding to up-to-date regulatory requirements.

2. GENERAL DESIGN

In the design of the V-407 reactor plant, more stringent criteria are applied than those in the requirements of regulatory documents, providing greater safety margins under all design conditions of NPP operation. Long operating experience with 16 power units with WWER-440 reactors and 36 power units with WWER-1000 reactors both in Russia and CIS countries, and in Finland, the Czech Republic, Slovakia, Bulgaria and Hungary, provided valuable input for the design of the V-407 reactor. A qualitative improvement of the WWER is achieved by the application of approved reliable systems, layouts and equipment and recording and removal of drawbacks revealed in operation of the WWER and PWR of the preceding generation.

Safety assurance is provided by a defence-in-depth approach. This is based on a system of barriers to prevent the spread of ionizing radiation and radioactive substances into the environment, as well as to a system of technical and organizational measures to protect and maintain the effectiveness of each barrier.

The system of barriers at the NPP with a V-407 reactor includes:

- a fuel matrix;
- fuel rod claddings;
- a boundary of coolant circuit cooling the core (the primary circuit); and
- double containment.

When the failure of any barriers or of the means of barrier protection is revealed during operation, the operation is stopped. Optimization of the technological processes under normal operation provides for low radiation impact to the environment; normal operational radiation in the plant surroundings amounts to around 0,1 per cent of the background. Under beyond design basis accidents accompanied by core melting, evacuation of the population outside the site boundary (radius 1,5 km) is not required. The probability of the occurrence of such an event is estimated to be of the order of 10^{-10} and assigned to the residual risk category.

In units with sealed safety barriers, a system of double containment is applied. The inner steel envelope is designed for full emergency pressure; the outer envelope, of reinforced concrete, is designed for external effects. All the equipment of the primary circuit and safety systems is inside the protective and sealed envelopes. The inner envelope forms a barrier to the spreading of radioactive medium under all design basis and beyond design basis accidents. The design pressure and temperature for the inner envelope are 0,7 MPa and 180 °C, respectively. Emergency dumping of the steam-air medium into the atmosphere through special filters prevents a pressure increase in the inner containment above permissible levels. Design untightness of the containment is less than 0,2 per cent of the volume per day. The outer envelope protects the sealed envelope and the reactor plant equipment against external impacts such as aircraft crashes, impact waves caused by explosions, hurricanes, tornadoes, etc.

The main equipment of the V-407 reactor plant, such as the reactor, main coolant pipeline, steam generator, pressurizer, ECCS hydrotanks, CPS drives, and the fuel handling machine, is similar to the standard-manufactured equipment operated at NPPs with reactor plants of the V-213 (WWER-440) and V-320 (WWER-1000) types.

Thefuel assemblies applied as nuclear fuel in the core of V-407 reactor plant are similar to the fuel assemblies of WWER-1000, which have shown high reliability indices through long-term operating experience. The specific loads in the core of the V-407 reactor plant are significantly lower, and very high fuel reliability is anticipated. The maximum linear heat rate on the fuel rods does not exceed the value of 26,5 kW/m.

3. SAFETY SYSTEMS

In the design of the NPP with WWER-640, the safety systems are based on passive principles of operation and inherent safety features, such as: self-control, heat capacity, heat conduction, heat transfer, gravitation, energy of compressed gas and other similar processes.

Passive systems are applied for emergency core cooling under design basis and beyond design basis accidents: the passive heat removal system (PHRS), emergency core cooling system (ECCS), system of primary circuit depressurization, and an active system of boron supply by high-pressure pumps into the reactor - under accidents of ATWS type.

The following safety systems are applied for the first time in the V-407 plant design:

- passive heat removal system (water PHRS).
- system of emergency depressurization of the primary circuit;
- system of low pressure ECCS tanks; and
- system of the reactor vessel cooling from outside.

The PHRS provides for heat removal from the containment and from the steam generators. It works on natural circulation due to differences in elevation between the heat source and the heat sink. The heat sink consists of tanks of chemically demineralized water, located on the outside of the protective concrete envelope and protected against external effects by concrete shielding. Heat removal from the steam generators takes place in the event of loss of all sources of power, including station-service diesel-generators, and under conditions of primary leaks. The PHRS begins by opening the starting valves on the pipelines connecting the water volumes of the steam generators with the heat exchangers in tanks of chemically demineralized water. The opening of the valves is automatic and does not require external power or control signals.

In the event of a primary system leak, the emergency depressurization system of the primary circuit forces a decrease in the primary system pressure. This depressurization ensures connection of the ECCS tanks to the reactor, and subsequently, long-term cooling of the core by natural circulation through the emergency pool. This task is performed by depressurizing valves on the lines connecting the primary circuit to the fuel-cooling pond, at the reactor inlet and outlet. The valves are actuated passively by a compressed spring acting on the valve stem; the spring opens the gate when the pressure in the primary circuit has decreased below a set value. The valves remain open until a forced closing signal is initiated.
After depressurization of the primary circuit, the tanks are automatically connected to the reactor.

Removal of the core residual heat is effected by water spillage from the tanks through the core, under gravity. Cooling water that passes through the core collects in the lower part of the sealed volume. In the final stage of an accident the sealed volume is filled with water up to a level above the elevation of the reactor "hot" nozzles. At the final stage of an accident, long-term heat removal from the core is effected by natural circulation in the pool.

The passive safety systems in the V-407 design makes it possible to meet strict safety criteria without applying active systems. This is valid for the whole spectrum of design basis accidents, without operator action, and in the event of loss of all AC power for a period of up to 24 hours. The V-407 reactor plant design includes technical means and organizational measures to prevent design basis accidents and mitigate their consequences. In addition, it incorporates technical means for keeping and cooling the corium in the reactor vessel.

The control and protection system (CPS) of the reactor comprises 121 absorber rods (ARs). The reactivity worth of the rods ensures that core sub-criticality will be attained at reactor scram, even in the event of sticking of the most effective control rod (CR). This is also valid under conditions of complete replacement of the boric acid solution by "pure" condensate at temperatures below 100°C and full xenon decay. Such improvement in the effectiveness of the emergency protection system yields a considerable safety margin where accident conditions combine with uncontrolled cool-down of the primary circuit and unauthorized injection of pure condensate. The control system and the core are constructed so the reactor will have a negative temperature coefficient of reactivity under any accident, and that core sub-criticality will be ensured without additional injection of boric acid. The neutron fluence to the reactor pressure vessel, by fast neutrons with an energy above. 0,5MeV, during the design service life of 60 years does not exceed the value of $2{,}7 \cdot 10^{19}$ neutrons/cm^2 . This is more than two times less than the neutron fluence to the vessel of WWER-1000 for 40 years.

Horizontal steam generators, similar to those in the PGV-440, are used in the V-407 reactor plant, as in traditional Russian NPP of the WWER design. The primary collectors in the steam generators of the V-407 reactor plant are stainless steel with an increased pitch between the heat exchanging tubes to reduce the possibility of crack formation during operation. In addition, the coolant pumps have a side coolant supply, making it possible to eliminate the loop seals in the main coolant pipeline. The main coolant pipelines are also stainless steel.

The V-407 reactor plant has adopted the "leak-before-break" (LBB) safety concept for design of the primary and secondary pipelines. Application of the LBB criteria is assumed to eliminate the risk of guillotine breaks of a main coolant pipe. A crack in a pipeline will normally be detected during periodic inspection, before it becomes a real leak, or the leak will be detected by the monitoring system, long before it approaches a critical dimension. Hence, installation of pipe whip restraints and guards can be largely eliminated. These measures have considerably improved the reliability and safety of the V-407 reactor plant. This can be illustrated by an example; the maximum temperature of the fuel rod cladding will not exceed 800°C in the event of a guillotine break of a main coolant pipe (MCP) with a nominal diameter of 620 mm, combined with a postulated station blackout for 24 hours.

In the analysis of beyond design basis accidents, calculations based on probability show that with the technical measures existing in the design, the probability of severe core damage is on the order of 10^{-9}-10^{-10}, i.e., this can properly be designated a residual risk. At the same time, for the hypothetical event of a core melt situation, it is possible to cool the reactor vessel from outside and prevent the corium from penetrating and slumping outside the reactor vessel.

Considerable improvement in the safety of the V-407 reactor plant has also been achieved by:

- improvement in the quality of equipment and systems at all stages of the life cycle of the reactor;
- implementation of "Quality Assurance Programs" at each stage of the NPP life cycle - from development to decommissioning, with strict observation and supervision of these programs;
- application of diagnostic systems for monitoring the technological parameters and state of reactor materials, steam generators and other equipment and pipelines both of the reactor plant, and of the systems important to safety.

The safety of the V-407 reactor plant has been considerably improved by providing the WWER-640 NPP power units with up-to-date I & C of improved reliability, and integrated self-diagnostics and expert systems to advise the operators of ongoing conditions. A simplified layout, flow diagrams and design of the V-407 reactor plant, removal of large heat exchanging equipment and a decreased number of pumps also contribute positively to safety.

The main characteristics of the V-407 reactor plant are presented in Table I.

TABLE I. TECHNICAL CHARACTERISTICS OF REACTOR PLANT V-407

	Characteristic	Value
	Reactor power: thermal/electrical, MW	1800/640
	Coolant flow rate through the reactor, m^3/h	53480
	Coolant pressure at the core outlet, MPa	15.7
	Average coolant temperature at the reactor inlet, °C	294.3
	Average coolant temperature at the reactor outlet, °C	322.7
	Pressure of generated steam at the SG steam header outlet, MPa	7.06
	Steam capacity of one SG, kg/s	248.4
	Feedwater temperature (under nominal conditions), °C	230
	Number of circulation loops	4
	Number of RCPs	4
	Total number of assemblies in the core,	163
	Number of assemblies with CPS rods,	121
	Design service life, years	50 (of vessel – 60)

4. List of Abbreviations

AR	absorber rod
CR	control rod
CPS	control and protection system
ECCS	emergency core cooling system
I & C	instrumentation and control
LBB	leak-before-break
MCP	main coolant pipeline
NPP	nuclear power plant
PHRS	passive heat removal system
RCP	reactor coolant pump
SG	steam generator
WWER	water-cooled, water-moderated power reactor

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DESIGN OF NPP OF NEW GENERATION BEING CONSTRUCTED AT THE NOVOVORONEZH NPP SITE

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Abstract

The design of a new generation NPP is described, underscoring advances in physical attributes and passive safety systems based on experiences with earlier designs at operating NPPs. This paper elaborates on systems for handling and storing radioactive wastes, on refinements in containment measures and on experimental and analytic validation of critical design factors.

1 GENERAL LAYOUT

The general layout of the nuclear power plant (NPP) was developed for a twin-unit station, considering the following requirements:

- maximum independence of each Unit;
- modular construction using the monoblock principle;
- optimal arrangement of buildings and structures of the main production process, as well as support production and auxiliary buildings and structures;
- mitigation of extreme external effects on NPP operability; and
- site zoning versus main production and auxiliary buildings.

The main building complex is in the center of the general layout (cf. Figure 1). It consists of the reactor and turbine plants, including outdoor main and standby transformers, standby power diesel units, spray ponds of the cooling system for reactor plant vital loads and a common sheltered center for management of emergency actions at the NPP.

2 INFORMATION ON THE MAIN PROCESS SOLUTIONS

2.1 Process solutions pertaining to the Main Building complex

2.1.1 General provisions and schematic solutions

The NPP with the ALWR class V-392 is a monoblock with a four-loop reactor and two turbine-driven feedwater pumps. The monoblock layout includes the double containment of the reactor plant, the turbine hall, the safety and auxiliary systems building and ensures a minimum length of engineering infrastructure lines, as well as high reliability in both normal operation and safety functions. The main engineering solutions of the design are aimed at:

- achieving a new qualitative level of safety compared to the V-320, and compliance with recommendations of the International Safety Group (INSAG)and the International Atomic Energy Agency (IAEA), etc.;
- application of mature processes and design solutions only (an evolutionary approach);
- improvement of economical performances compared to theV-320 and fossil fuel sources;

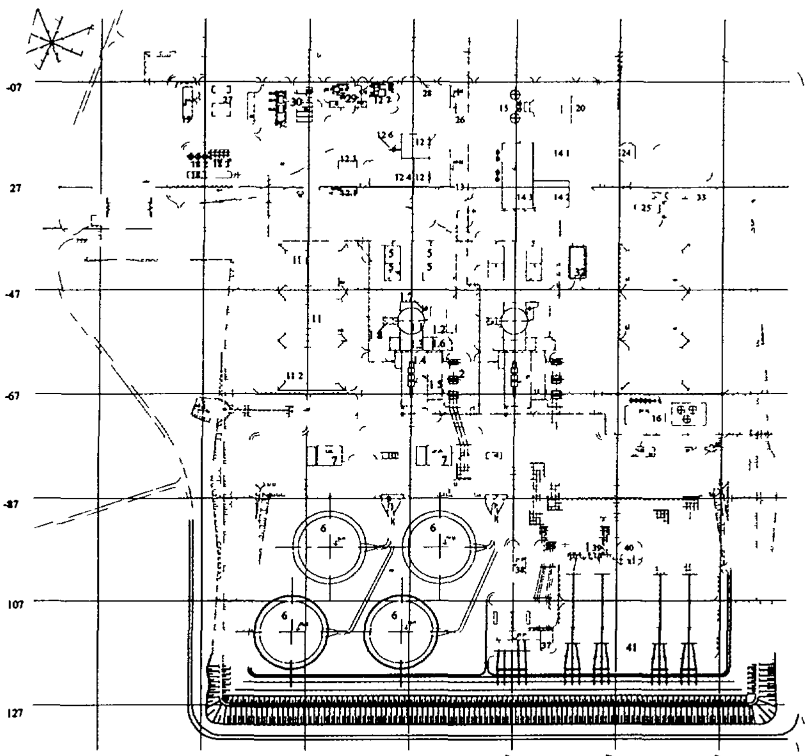
- application of feedback from operating NPPs and results of analysis by domestic and foreign companies; and
- creation of prerequisites and scope of preparatory works for implementation of a large power unit by the year 2020, with mature inherent safety features, intended for large-scale application in Russia. It features obvious advantages in efficiency compared to fossil fuel plants, irrespective of the region in which it is to be located.

The process solutions are essentially based upon:

- an advanced VVER-1000 with an improved reactor trip system, capable of maintaining reactor sub-criticality during cooling down to a temperature of 100 - 120 ^{0}C without boron injection; reactivity feedback is improved by negative coolant temperature coefficients of reactivity throughout the fuel cycle;
- advanced steam generators with a modified primary header structure using austenitic stainless steel in the heat exchanger tubes to extend service life; blow-down is arranged from the section with the highest salt concentration in the steam generator boiler water;
- an advanced reactor coolant pump with a shaft seal that prevents the coolant from leaking in case of a loss of power for 24 hours and loss of sealing water and other cooling media;
- auxiliary systems of reactor and turbine islands. The operating experience of many units is used during design development; new engineering solutions tested at operating units are used as well as an approach aimed at continuous diagnosis of disturbances.
- radioactive waste processing and storage systems using new advanced process solutions alongside traditional ones, with efficiency proven by many years of experience at domestic and foreign NPPs. Selection of new processes and new equipment is justified by research and development (R&D), and orders are placed with manufacturers to develop new equipment.

2.1.2 Schematic solutions

The reactor plant comprises four steam generator loops. The primary coolant temperature is 322^{0}C at the reactor outlet, and the design primary pressure is 17,6 MPa. Each loop has one reactor



Legend for buildings and structures

No on Drwg.	CCS code	Designation
1		Main building complex
1.1	UJA	Containment and Process Safety Systems Building
1.2	UKC	Normal Operation Process Bldg. of Active Water Treatment System
1.3	UBP	Control Safety System Bldg.
1.4	UMA	Turbine Generator and Deaerator Bldg. incl. Lub. oil facilities
1.5	UBA	Normal Operation Electric Device Bldg. and Ventilation Centre
1.6	UYB	Unit Sanitary and Welfare Building
1.7	UKH	Ventilation stack
1.8	UJG	Transportation bridge
1.9	UCA	Monitoring and Control Systems Bldg.
2	UBF	Transformers with transportation tracks
5	UKD	Standby Diesel Power Plant Safety Systems
6	URA	Cooling Tower
7	URD	Pump station for cooling tower with back-up tank
8	URJ	
11	URR	Spray pool
12		Set of auxiliary structures of the controlled area
12.1	UFC	Fresh fuel storage with emulator room
12.2	UKS	Solid waste treatment Bldg. with storage Cooling towers discharge channels and integrated structures
12.3	UFA	Spent fuel container storage
12.4	UKU	Repair shops of controlled area
12.5	UKW	Laboratory and welfare bldg. and active laundry
12.6	UKH	Ventilation stack with air duct for CRS of controlled area
12.7	UCY	Biological and deep treatment station for waste water from controlled area
14		Set of auxiliary structures of the free access area Common auxiliary building
14.1	USY	Control repair shop and control motors store
14.2	UYC	Administration and Laboratory bldg. and Canteen
14.3	UGD	Chemical water treatment with tank facilities
18	UTM	Lub. oil and diesel facilities
25	UVE	Gate house
27		Production and fire water supply structures
28		Domestic waste pump station of the free access zone
29		Sewage piping of drains water contaminated with petroleum products
30		Sewage piping of production and rain drains water
32	O1U2M	Protected central control station
33		Parking lot
41		Open-air switchgear for 500 kV

FIG. 1 General layout of NV NPP-2 [Novovoronezh NPP-2]

coolant pump (RCP) with external motor and necessary inertia performance and one horizontal steam generator with submerged heat exchanging surface. The live steam pressure is 6,37 MPa (design pressure is 8,0 MPa (80 bar)), and the steam capacity of the plant as a whole is ≈ 4 x 1470 t/h.

The reactor plant is equipped with four, first-stage accumulators with a nitrogen cushion pressure of 6,0 MPa (60 bar). The first-stage accumulators are connected in pairs through check valves to the emergency nozzles in the upper and lower plenums of the reactor pressure vessel. The bypass line of each RCP is equipped with systems for high-temperature mechanical cleaning of the coolant. Tanks with high concentration boric acid and quick-acting valves of the boron supply system are connected to the bypass lines, backing up the function of the solid absorbers of the reactor scram system.

The 1000 MW power turbine plant has an optimized steam cycle with regenerative heating of the feedwater. The feedwater plant consists of two non-redundant, turbine-driven feedwater pumps. It is capable of maintaining 70 per cent of the unit steam capacity with one pump operating.

The process part of the unit safety system is a train-type structure to fulfil the criteria of high reliability of crucial safety functions and minimize common-cause failure probability. This approach led to safety systems design with mutually redundant active and passive trains; this diversity covers practically all the main safety functions. Safety systems solutions are described in more detail in the following sections

2.1.3 Nuclear Fuel Management

The nuclear fuel management system facilitates all fuel handling at the NPP and comprises the following systems:

- fresh fuel storage and management system, including fuel transfer to the reactor department;
- the refueling system;
- a spent fuel handling system consisting of,
  - spent fuel storage near the reactor, and
  - spent fuel storage in a special building outside the reactor building;
- a system of nuclear fuel transportation at the NPP site covering operations from reception of a special carriage with fresh fuel, to sending away a special carriage with spent fuel and internal site transportation of nuclear fuel.

Figure 2 depicts the movement of fuel to, within and away from the NPP.

2.1.3.1 Fuel Handling in the Fresh Fuel Storage Facility (FFSF)

The FFSF is common to the whole NPP and is in a separate building at the site. For nuclear safety, the FFSF is assigned to seismic category 1, which means that FFSF structures and the fresh fuel handling equipment are designed to withstand extreme external effects. FFSF is designed to house:

- 170 fuel assemblies necessary to refuel 2 reactors, plus 20 per cent;
- 180 fuel assemblies in casks for complete core loading plus 10 per cent.

FFSF is equipped with racks to store the fuel assemblies and absorbing rods prepared for refueling. The rack is a metal work consisting of three slabs rigidly connected by pillars. The cells housing fuel assemblies are arranged in a 400-mm, triangular array. Before refueling, the prepared fuel assemblies are installed by the FFSF into a site-internal transportation container, which in turn is mounted onto a site internal platform and transported to the reactor department.

2.1.3.2 Fuel Handling in the Reactor Island

The main fuel handling operations conducted within the NPP reactor island comprise:

- fresh fuel delivery into the reactor island and loading into the reactor;
- spent fuel removal from the reactor;
- spent fuel storage in the spent fuel pool for not less than 3 years; and
- cooled fuel removal from the reactor island.

Cooled fuel from the spent fuel pool is removed simultaneously with the preparation for refueling operations on the reactor. Core refueling covers replacement of spent fuel assemblies and spent burnable absorber rods by new ones and fuel assemblies and absorber rods shifting in the core. A refueling machine handles fuel assemblies and absorber rods under a protective water layer. During these operations, the refueling machine can handle only one fuel assembly or one burnable absorber rods bundle at a time; fuel assemblies may be shifted together with absorber rod bundles.

After removal of cooled fuel from the reactor island, withdrawal of spent fuel out of the reactor and rearrangement of fuel assemblies within the core, the fresh fuel prepared in the NFSF and delivered to the reactor island is loaded into the core. The spent fuel unloaded from the reactor is stored in spent fuel pool compact storage racks consisting of borated steel cells. Fuel assemblies in the racks are stored in a 300-mm triangular array. Fuel in the spent fuel pool is protected by water with a boric acid concentration of 16 g/kg.

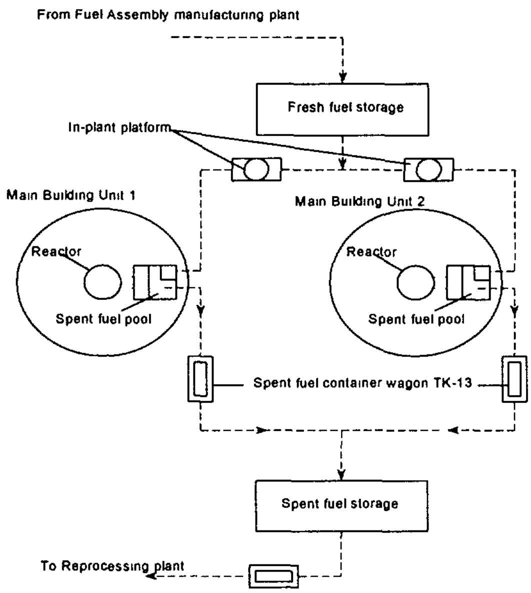


FIG. 2 General fuel movements at NPP

2.1.4 Radioactive Waste Management

2.1.4.1 Radioactive concentrated liquid media processing

The process for treating liquid radioactive effluents includes:

- collection and temporary storage; and
- processing and solidification in concrete.

Liquid radioactive waste is stored separately depending upon composition and activity level. Temporary storage is envisaged for waste accumulation to allow decay of short-lived nuclides prior to processing in the solidification plant. Medium-active solutions are stored for 3 months, while dry residues and low-active solutions are processed as accumulated. The system is designed to solidify liquid waste in proper containers for transportation and safe long-term storage in the processing and storage building. The following technologies are used for liquid waste processing, depending upon composition and activity;

- concentration of low-active dry residues to obtain dry salts; and
- concrete solidification of medium-active ion exchange resins, sludge, and salt concentrates.

The method of waste storage enables extraction of waste from storage cells to check packaging or for removal to regional disposal.

2.1.4.2 System of Solid Radioactive Waste Processing and Storage

The design of solid and solidified radwaste processing and storage is based on process needs, observing safety concerns during normal plant operation and during design basis accidents. With respect to nuclear safety, solid radwaste processing and storage applies a system of barriers to the potential propagation route for radioactive substances to the environment.

2.1.4.3 System of Solid Radwaste Collection, Sorting and Transportation

Solid radwaste is collected and sorted where it is generated with regard to level of radioactivity and processing methods and loaded into appropriate containers or disposable packing material. Containers and disposable packing material are brought to unattended rooms when repair work with waste generation is to be performed; in rooms that are periodically attended and rooms having permanent personnel, containers are installed in allocated positions. The number and types of containers are determined in advance by prediction of waste quantity, composition and radioactivity. Containers filled with waste are combined into lots for removal to the processing and storing building. Special areas are provided in the main building complex for this purpose.

Special vehicles remove the waste from the reactor island through the transportation corridors. Before leaving the transportation corridor, vehicles are subject to dose metering, washed and decontaminated, if necessary. Solid radwaste from the containment is removed through the equipment lock.

2.1.4.4 Solid Radioactive Waste Processing System

To reduce the volume of solid radwaste to be stored, it is processed by the following methods:

- grinding;
- incineration; and
- compacting.

The final product of processing is packed into standard drums. Metal waste (pipes, rolled stock, etc.), ventilation filters, and small items of equipment are ground. Solid radwaste of the 1st and 2nd radioactivity group is compacted. Combustible waste of the 1st and 2nd radioactivity groups is incinerated. Solid radwaste processing and storage facilities are in a separate building.

2.1.4.5 System of Solidified Liquid and Solid Radwaste Storage

A solid and solidified-liquid radioactive waste storage facility is provided at the site. Solid radwaste is stored in a specially equipped above-grade reinforced concrete storage building with walls and ceilings sufficiently thick to ensure mechanical strength and biological shielding. The waste storage method enables withdrawal of waste from storage facility cells for package inspection or for removal and transport to the regional repository. Solid radwaste and solidified products are placed in the storage facility in baskets containing 6 standard drums, each for subsequent waste withdrawal and transport to regional disposal. The filling coefficient of the cells in case of such storage is 0,34.

Special drums with highly radioactive waste are stored along guide lines in the storage cells for 50 years (the service life of the NPP), with the feasibility of their subsequent removal to the regional disposal. The filling factor is in this case 0,59.

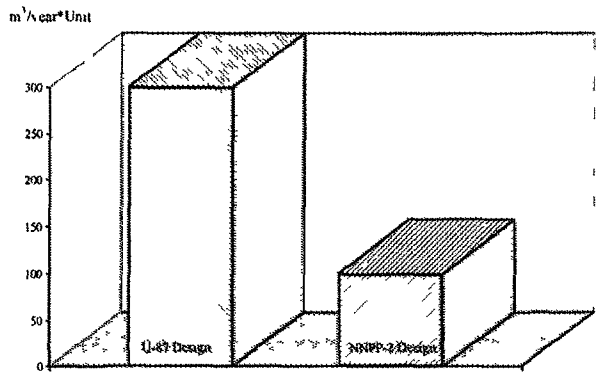
Nowadays, the concept of solid radwaste storage at the site for the whole NPP service life of 50 years is adopted. Nevertheless, the technology and storage structures enable withdrawal of drummed waste for further processing and subsequent transport to the regional disposal, as soon as it has been designed and constructed. The solid radwaste storage facility at the NPP site is built in stages with subsequent extension. The initial storage volume is designed for 10 years and is commissioned together with the NPP pilot unit; subsequent extensions are carried out, if required.

The storage facility is equipped with railroad and motor vehicle access points, systems of inspection and decontamination of transport, a radiation monitoring system, a system of explosive and fire hazard detection, and a heat and humidity detection system.

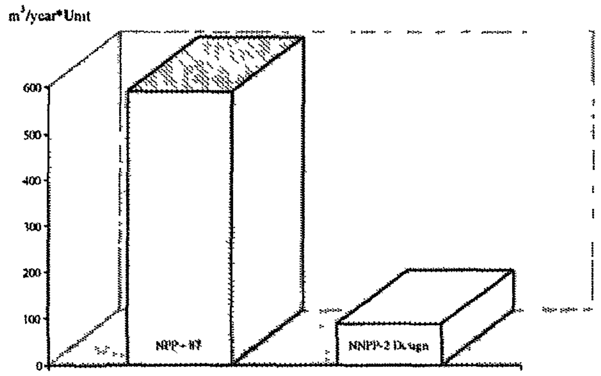
The generation of radioactive waste has been reduced considerably compared with that in existing plants. Figures 3 a and b illustrate the improvement by comparing the annual volumes of evaporator residue and solid radwaste from the NV NPP-2 with those for the U-87 design.

2.2 Main Engineering Solutions Pertaining to the Electrical Part

An overview of the electrical systems is shown in the single-line diagram of Figure 4. High-voltage (HV) switchgears for 500 and 220 kV are provided in the design for NPP power output to the grid.



a) Evaporation residues per year



b) Solid radioactive waste per year

FIG. 3 A comparison of radwaste volumes at NV NPP-2 and U-87

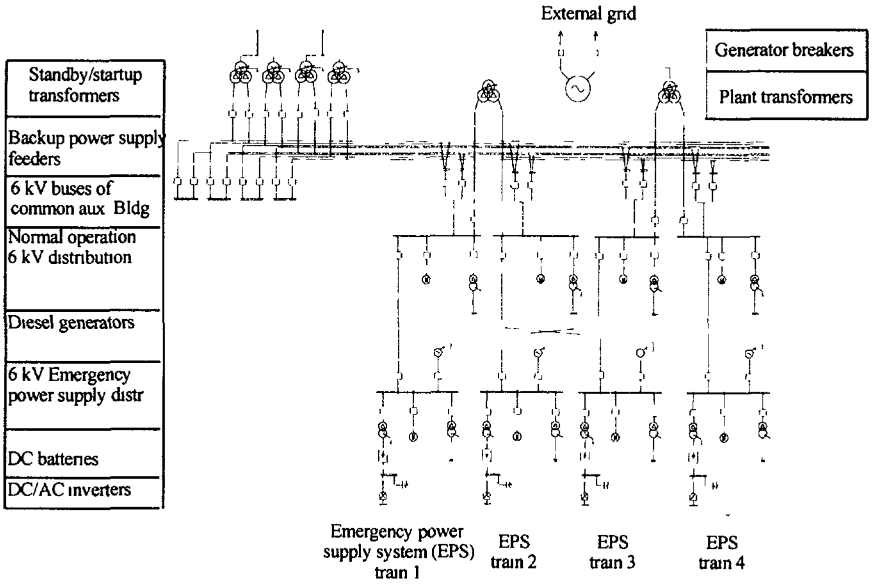


FIG. 4 Basic single-line diagram

An outdoor 500 kV switchgear is constructed in the 500 kV line; a metal-clad SF_6 gas-insulated switchgear (MCS) is provided for the 220 kV line instead of an outdoor switchgear due to the vicinity to the cooling tower and the reduction in land.

According to the design, each unit is equipped with one completely water-cooled turbine generator of 1100 MW. The generator and the main transformer are connected by shielded bus ducts with generator breaker units capable of breaking short-circuit currents. Two normal operation plant transformers of 63 MVA each are installed in the tap between the generator breaker and the main transformer.

The availability of a generator breaker enables unit start-up and shutdown with power supplied from the grid through the main transformer. It also enables continued application of auxiliary power from the grid via normal operation plant transformers in the event of the failure of a generator or process part of the unit without changeover to a standby transformer. This enhances the unit reliability considerably.

The auxiliary power supply system of the NPP contains power sources for normal operation, standby and emergency power supply. The auxiliary power supplies are divided into off-site and internal. The power grid with its power plants represents the off-site power supply. Off-site power may be supplied to NPP auxiliaries through the normal operation plant transformers or through the standby transformers 220/6,3-6,3 kV. The internal normal operation auxiliary power sources are the turbine generators and emergency auxiliary power sources – diesel generators and storage batteries. The auxiliary power system is designed to supply loads supporting:

- NPP normal operation;
- bringing the Unit into safe condition and maintaining it in normal and emergency conditions; and
- state of the reactor plant monitoring for 24 hours in case of loss of power and failure to start all the diesel generators.

Normal operation and emergency auxiliary power supply systems are envisaged at the Unit. Each Unit is equipped with two working transformers of 63 MVA each, feeding normal operation and emergency power supply system loads under normal plant operation. The design envisages four normal operating sections of 6 kV, in accordance with the number of RCPs. Power to each RCP is supplied from individual sections so the Unit operation is more stable in case of the loss of 6 kV sections. Loss of an RCP requires Unit power reduction or shutdown.

The emergency power supply system consists of two independent subsystems; each of the independent subsystems consists in turn of two trains with mutual redundancy. Installation of diesel-generators of 6300 kW and startup time of 1,5 s and three storage batteries is envisaged in each train as power supplies. Emergency power supply system switchgears are connected to normal operation loads ensuring serviceability of the main process equipment, which requires power in case of a loss of normal supply. Turbine oil pumps, shaft turning gear and rotor hydro-jack are examples in this category. Normal operation loads are connected to the switchgear of only one independent subsystem.

Emergency power supply system storage batteries are intended:

- to provide power to control and automation and relay protection devices of emergency power supply system elements and emergency lighting of loads of this channel of the emergency power supply system through one storage battery. Battery discharge time is 2 hours.
- to provide power to I&C hardware through the second battery. Duration of battery discharge is also 2 hours;
- to provide power to reactor control and monitoring devices in case of total loss of AC power through the third battery. Battery discharge time is 24 hours.

The electrical equipment of the emergency power supply system is located in the standby diesel power station (SDPS). The SDPS for each subsystem are in two separate buildings. Each building consists of two physically separated cells and each cell houses the equipment of one train. The trains are separated by structures with a fire resistance of at least 1.5 hours.

2.3 Computer-aided Process Control System (Instrumentation and Control[I&C])

Each NPP unit is envisaged as having independent I&C systems. From the unit point of view, the I&C is intended to maintain design basis limits set by process parameter values and characteristics of state of process elements and systems designed for normal operation conditions, operational occurrences, emergency situations and accidents. In developing the design, consideration has been given to comments by the IAEA on the automation of power plant units with RU-320 and their operational experience. The I&C hardware was developed in parallel with the plant design. The I&C concept is based on the following fundamentals:

- I&C is mainly implemented using modern digital hardware proved by positive operating experience at fossil fuel or nuclear power plants;
- the design adopts a centralized system of unit equipment control, which envisages unit automatic control from the main control room;
- a standby control room is provided at the unit to ensure unit shutdown, cooling down and reactor plant sub-criticality monitoring;
- a protection system train is envisaged for each of the four safety system trains;
- two independent sets of scram (emergency protection) are envisaged for emergency shutdown; and
- a unit top level system is envisaged to combine all the automation sub-systems into a unified system, which implements unit common tasks as well.

The structural I&C diagram (cf. Figure 5) of the unit represents the main components of the system/protection systems, low level automation and unit top level control and monitoring systems, as well as digital trains of data exchange and remote control.

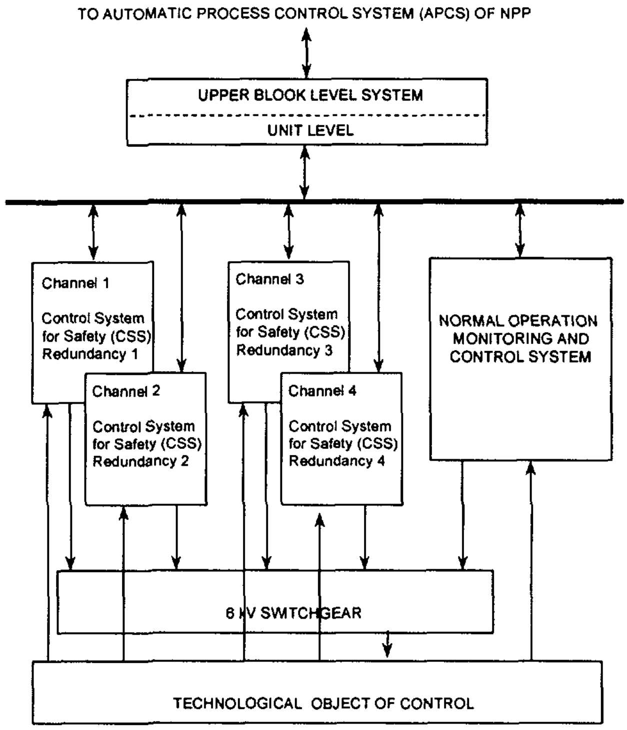


FIG. 5 Structural diagram of the automatic process control system (APCS)

2.4 Civil Solutions, Main Building Complex General Layout

The main building complex layout is of monoblock type; the main and ancillary equipment of the reactor and turbine plants of each unit are located in separate compartments. The main building complex, which is a standardized module, combines the reactor and turbine islands and a sanitary-social building.

2.4.1 Reactor island

The reactor island is a number of adjacent compartments containing the reactor plant and systems supporting normal operation and those ensuring emergency shutdown of the unit. The reactor island consists of the containment building and process safety systems, buildings of the protection safety systems, monitoring and control systems building, normal operation process and special water treatment systems building. All buildings are constructed on separate foundation slabs. To achieve independent response under static and special dynamic effects, gaps between buildings are assumed to be 400 mm.

2.4.2 Containment and process safety systems building

The containment building consists of a cylindrical containment (the leaktight part) and two adjacent buildings on opposite sides (non-leaktight parts), on the same foundation slab. The containment houses safety-related systems, and therefore is designed to withstand external effects; it is in Safety Class 1.

The building plan dimensions are 73,2 x 43,2 m. The building height is 89,4 m. The containment is an accident confinement system and consists of two shells: an inner leaktight containment and an outer one protecting it from external effects. The reactor plant spent fuel pool, ancillary process systems working at primary parameters, ventilation systems and equipment providing for fuel handling and repair operations are located within the containment.

The reactor occupies the central part of the containment. The spent fuel pool and internals inspection wells, two Main Circulation Circuit (MCC) compartments housing steam generators, RCPs, Main Circulation Pipelines, pressurizer, bubbler and quick boron supply system tanks are located on both sides of the reactor pit. The Reactor Pressure Vessel inspection machine and the core catcher for beyond design basis accidents are located on the slab of the leaktight containment under the reactor.

Maintenance rooms for pulse valves actuators, purification systems rooms (SWT-1, SWT-2), control chambers for valves in contaminated pipelines and ventilation plants are located around the reactor pit. The ECCS tanks of the 1st and 2nd stages are located at the maintenance elevation. The contaminated equipment-washing unit is located next to the spent fuel pool from the side of equipment lock. For execution of transportation operations through the confinement boundary, the containment is equipped with air locks designed to ensure tightness under design basis accidents and external design impacts.

Personnel access into the containment is through the main air lock to the maintenance elevation from the controlled access area of the auxiliary process systems building. An emergency air lock is provided at the lower elevation to ensure the emergency exit of personnel from the containment. All fuel and equipment handling operations are carried out through an equipment lock at the maintenance elevation and ledge located outside the containment.

The containment building basement is the space between the leaktight containment slab and the building foundation slab. The structural features of the cooling down pumps define the height of the basement. The containment basement houses primary circuit and spent fuel pool cooling systems, primary circuit I&C and radiation monitoring assemblies (immediately under the core), the intermediate circuit system, the steam generators emergency cooling down and blow-down systems, I&C and radiation monitoring and I&C rooms of the systems located in the basement.

Filters of the containment overpressure release system, emergency inventory tanks for service water and exhaust ventilation center are in the building adjacent to the containment, on the side opposite the turbine building. The building in question is in the controlled access area, together with the basement.

The adjacent building from the side of the turbine hall, except for the floor of the rooms for the steam generators emergency cooling down and blow-down systems, belongs to the free access area. It houses the safety-related part of the main steam and feedwater pipelines, plenum ventilation center, cable corridors, I&C and radiation monitoring hardware premises.

Systems in the non-leaktight part of containment building (except for the containment overpressure release system) are divided into two independent channels with 100 per cent redundancy in each channel. Walls separate rooms of different channels. Areas of one channel (including rooms, corridors, and stairs) are completely isolated from the areas of another channel and no communication lines (pipelines, cable and ventilation ducts) of another channel pass through it. A vestibule separates corridors of the area of one channel from the common corridor to provide fire protection. Common corridor and emergency exits into the outdoor area provide for the evacuation of personnel.

2.4.2.1 Main Solutions Pertaining to Containment Civil Structures

Outer Containment

The outer containment is a cylinder with a spherical dome of monolithic reinforced concrete. The wall thickness of the cylindrical part and the dome is 600 mm and the inner diameter, 50,8 m. The outer containment will absorb loads of external impacts arising from hurricanes, tornadoes, external shock wave, and aircraft crashes. An inter-containment gap of 2,2 m enables maintenance of the inner containment pre-stressing system and accessibility for visual inspection of surfaces. This gap enables controlled collection of gas-air media leaks. The inner surface of the outer containment will be provided with polymeric coating ensuring the required tightness of outer containment. The heat exchangers of the passive heat removal system are located on the outer containment and the installation is arranged such that the heat exchangers of only one steam generator could be damaged in case of an aircraft crash.

Inner Containment

The inner containment is cylindrical pre-stressed monolithic reinforced concrete covered with a semi-spherical dome. The equipment layout in the leaktight space defines the basic dimensions of the containment, which are:

- cylinder and dome inner diameter, 44 m;
- cylindrical part height, 40 m;
- wall and dome thickness is 1,2 m, based on structural requirements and biological shielding.

A steel lining ensures leaktightness of the inner containment; basic ambient design parameters are:

- During design basis accidents:
  - emergency design overpressure 0,4 MPa
  - emergency design temperature + 150 ^{0}C.
- During beyond design basis accidents:
  - emergency overpressure is 0,6 MPa
  - emergency temperature is + 200 ^{0}C

2.4.3 Protection Safety Systems Building and Monitoring and Control Systems Building

The electrical building is located between the reactor and turbine buildings. There are two protection safety system buildings – a standard layout solution – that are in Safety Class 1. They are on opposite sides of the monitoring and control systems building to prevent simultaneous destruction in the event of an aircraft crash. Each of the buildings houses electrical equipment of two independent channels of active safety systems, including independent systems of plenum and exhaust ventilation.

The Safety Class 1 building of the monitoring and control systems electrical equipment is between the protection safety system buildings. It houses the main control room, control and protection system (SUZ) and information computer system panel. These buildings are prefabricated-monolithic reinforced concrete.

2.4.4 Building Normal Operation Process Systems of Special Water Treatment System (SWT)

The dimensions of the process systems (PB) and special water treatment (SWT) building are 45 x 66 m (Safety Class II). It is a process support to the containment building and adjoins the latter by the long wall from the side of the transportation ledge. The building houses auxiliary primary systems (PB) including Unit special water treatment. The building is monolithic reinforced concrete. Special sewage, borated and service water drainage collection systems are at lower elevations. Laboratories are designed at elevation 7.200. I&C, control and protection system premises, plenum ventilation center, exhaust ventilation center and stack are at higher elevations. An access air lock to the containment leaktight space is at elevation 31.800; entrance into the building is through the sanitary-social service building.

2.4.5 Turbine Hall and Deaerator Rack Including Oil Handling Building

The layout of the turbine hall, deaerator rack and oil handling building is determined mainly by the type of turbine plant, with condensers in the basement, three low-pressure cylinders and a water-cooled generator, as well as upon the layout of auxiliary systems and equipment. The turbine hall and deaerator rack, including the oil handling building are in Safety category II. The turbine hall dimensions are 36 x 102 m, with a height of 40,8 m; the turbine hall is a steel structure. Placing the turbine hall with its end facing towards the reactor enables the best use of the layout volume for the arrangement of equipment and to locate turbine steam exhausts as close to the reactor building as possible.

The turbine plant foundation is provided with vibration dampers so that the transfer of dynamic and vibrational effects on civil structures of platforms and ceilings, resting on the turbine plant foundation pillars and the lower support plate, are practically avoided.

2.4.6 Standby Diesel Power Station (SDPS)

The SDPS (Safety Class 1) is intended to supply power to safety system loads under loss of off-site power to the plant. The equipment of each safety system train is located in an isolated SDPS cell. Four cells with dimensions of 30 x 33 m and 16 m high are provided for each unit. The cells are arranged in pairs, with their short sides contacting each other, and they are located in two separate buildings, to prevent their simultaneous destruction in case of an aircraft crash. Each cell houses:

- the standby diesel power plant itself;
- intermediate circuit and vital consumer service water supply pump house.

The building is a monolithic reinforced concrete structure.

2.4.7 Sheltered Center of Emergency Actions Management at NPP (SEAMC)

The SEAMC is designed as a sheltered two-story underground structure belonging to Safety Class 1 and seismic category 1. Its plan dimensions are 24 x 54 m. The building houses:

- standby control rooms of Units 1 and 2 (SCR1 and SCR2);
- NPP central control room (CCR); and
- shelter for 900 persons.

The SCR is intended to shut down the reactor unit in the event of an MCR failure. From the SCR, it is possible to monitor and initiate the safety systems and remove heat from reactor plant. The status of the reactor plant and spent fuel pool can be monitored from the SCR under all operating conditions, including station blackout. SCR availability is ensured for 24 hours in case of loss of power by means of separate storage batteries. A system for recording crucial parameters ("black box"), ensuring information preservation in case of an accident at the plant unit, is located at the SCR. The NPP CCR is intended for control of the power generation process and NPP-common facilities, and radiation monitoring at the site and at buffer area. (ARSMS).

3 SAFETY CONCEPT, SAFETY EVALUATION RESULTS, RESEARCH PERFORMED

3.1 General Philosophy and Safety Concept

Unit 1 of the NVNPP-2 is designed as an NPP with the enhanced safety VVER-1000 reactor of a new generation. The safety concept has been elaborated in an evolutionary approach based on thorough analysis of operating experience and design solutions of NPP units with V-320. The analyses that incorporated an evaluation of advantages and weak points of these operating NPPs, were carried

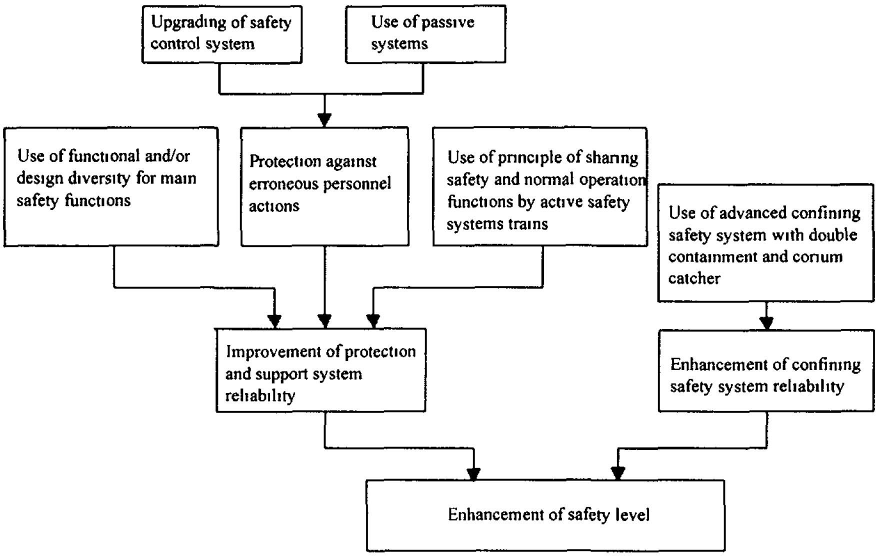


FIG. 6 Enhancement of the safety level in NPP with V-392

out within the framework of an effort to enhance the safety of these units. This effort was implemented in line with domestic and international programs with participation of leading Russian (AEP, OKB Gidropress, RSC "Kurchatov Institute" etc.) and foreign (EdF, GRS, Siemens) companies as well as the IAEA. The safety enhancement concept of the NPP with V-320 was developed using deterministic and probabilistic safety analyses. Based on the analyses, conclusions were drawn that the safety level of the operating NP units with V-320 reactors in major part complies with the safety level of other operating NPP units equipped with PWR reactors. Implementation of measures suggested for the NPP with V-320 safety enhancement concept would ensure compliance of these units with most of the requirements of the regulatory documents. However, the new qualitative level of safety can be achieved by elaboration of new design solutions ensuring resolution or reduction of the weak points revealed at NPP with V-320. It is necessary to point out that recommendations for the NPPs with V-320 were thoroughly considered for NPP with V-392 and were subject to special analysis focused upon two main points.

The safety concept of the NVNPP-2 is based on application to the maximum possible extent of the engineering principles of defense-in-depth concepts described in IAEA documents and on the input of data from operating NPPs to the V-320 safety analysis. The NVNPP-2 concept for safety enhancement that is schematically summarized in Figure 6, covers the following basic principles:

1. Application of functional and/or structural diversity in the systems performing each individual safety function. Mutually redundant active and passive systems are used in the design. Diversity ensures sufficient depth of protection against common cause failures and reduces safety systems unavailability indices by several decades.

2. Use of active safety system trains (emergency cooling down and ECCS) for execution of normal operational functions. At the same time, most of those components are in the states similar to the state they are in during the execution of assigned safety functions in the course of accidents. Such a mode of functioning of those systems makes it possible to enhance availability and ensure additional protection against common cause failures. For continuously functioning components, latent failures, the main cause of unavailability of the systems in the "waiting" mode of operation, are avoided.

3. The following design solutions provide protection against human error:

   •increased automatic system control (prevention of personnel interference) in case of a number of design basis accidents and, in particular, in case of primary to secondary leaks;

   •introduction of passive systems not requiring personnel actions for actuation.

4. Application of a full pressure double containment equipped with a hydrogen removal system, containment air discharge and purification (filter) system and core melt catcher, ensuring that the system does not exceed the established limiting release under beyond design basis accidents with severe core damage.

An overview of safety functions together with a list of systems capable of fulfilling each of them is provided in Table 1.
One should note that most design solutions mentioned above were developed based upon results of PSAs performed for operating NPPs with V-320 reactors and as a part of the design of NVNPP-2 Unit 1.

3.2 Safety Systems Technological Bases

The greatest modifications to the V-320 concern the safety systems. Probabilistic analysis revealed the dominant safety functions and "weak" points of the existing design and concluded that the following solutions are necessary:

- the main critical safety functions shall be fulfilled by diverse systems, both active and passive;
- in terms of functional reliability and taking into consideration peculiarities in the maintenance procedures, the best active safety system part structure is 4 x 100 per cent, at the same time the structure of 2 x 200 per cent yields the best results for support and protection safety systems.

Table 1. Structure and composition of the safety systems

Active part of Safety systems	**Structure**			
	Sub-system 1		Sub-system 2	
	Train 1	Train 2	Train 3	Train 4
Protection and Isolation systems				
- Multi-functional system of emergency heat removal from the reactor core	100 %	100 %	100 %	100 %
- Emergency heat removal system through the secondary circuit	100 %	100 %	100 %	100 %
Power supply systems				
- *Intermediate circuit and service water*				
▪ under closed conditions	100 %	100 %	100 %	100 %
▪ under open conditions	200 %		200 %	
- Ventilation	100 %	100 %	100 %	100 %
- Valves fail-safe supply				
• Redundant valves of the active part	100 %	100 %	100 %	100 %
• Isolation valves and passive part valves	200 %		200 %	
- *Pumps fail-safe power supply*	100 %	100 %	100 %	100 %
- APCS fail-safe power supply				
• Control safety system of the active part	100 %	100 %	100 %	100 %
• Control safety system of the passive part	200 %		200 %	
Control safety systems				
- Scram (emergency protection) system	200 %			
- Sensors	400 %		400 %	
- Logic part	100 %	100 %	100 %	100 %

Passive part of Safety systems	***Structure***
- Fast boron injection system	4x25 %
- Hydraulic accumulators of stage 1	4x50 %
- *Hydraulic accumulators of stage 2*	4x33 %
- Passive heat removal system	4x33 %
- Medium let-down and purification system from the containment	100 %
- Core catcher	100 %

The greatest reliability in main safety functions execution by active systems is achieved when the so-called "combination" principle is implemented, when active safety mechanisms perform normal operational functions and upon the appearance of accident indications, perform safety functions, either without change-over or with a minimum number of change-overs. Compared to traditional "waiting" safety systems, this solution improves reliability considerably (5-6 times). This is due to low sensitivity to latent failures (failures not evident in the "waiting" state of mechanism, which is out of operation) and reduces significantly the equipment (valves, cables, instruments, automatic devices, etc.). For example, in the NPP-392 design, there are four pumps in the active emergency cooling system through the primary circuit, whereas the same functions in the traditional design are carried out by 12 pump assemblies (The comparison refers to identical structures of 4 x 100 per cent).

The active part of the safety systems includes:

- scheduled and emergency, primary circuit and spent fuel pool cooling down ;
- steam generator emergency cooling down and blow-down;
- intermediate circuit system;
- service water supply system;
- ventilation and air conditioning support systems.

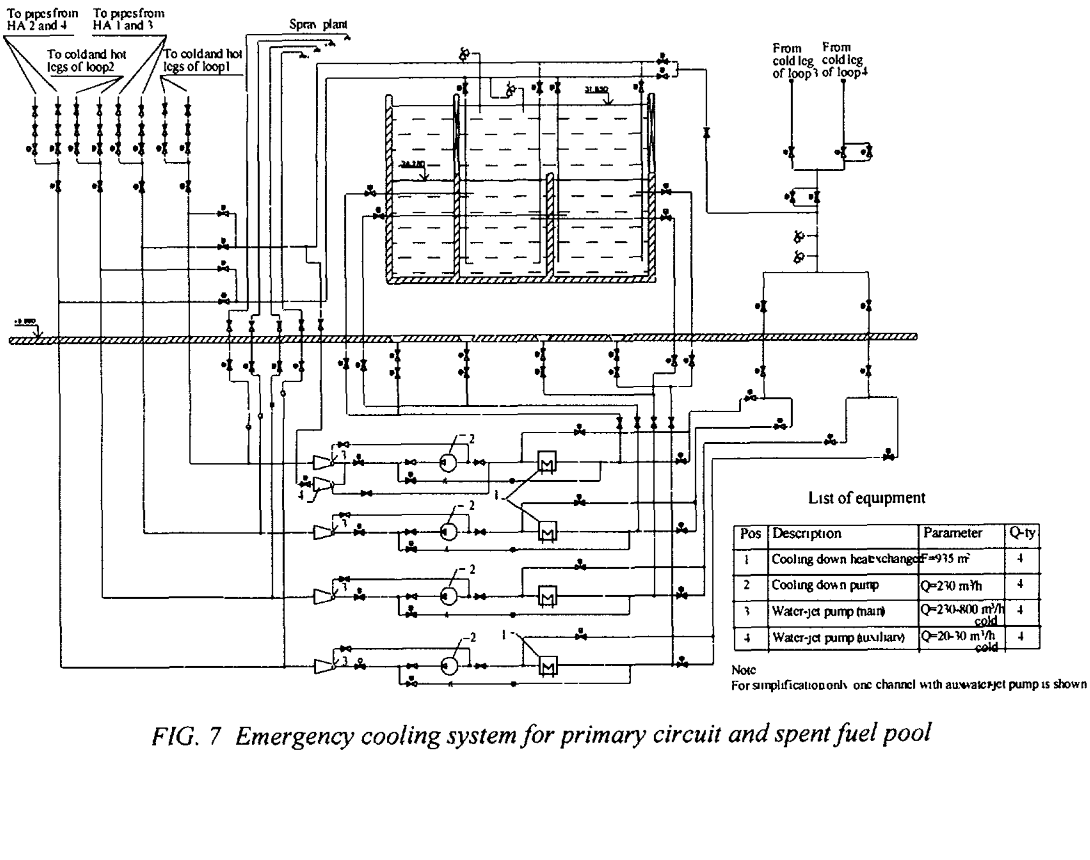
The passive part of safety systems incorporates:

- the passive heat removal system (PHRS);
- the 1st and 2nd stage accumulator system
- the quick boron injection system; and
- a system for maintaining under-pressure in the inter-containment gap.

The active emergency cooling system (cf. Figure 7) through the primary circuit comprises four groups of cool-down circuits with a combination of centrifugal and jet pumps in each. Under normal operation, these circuits are used for spent fuel pool cooling; under accident conditions, the system executes circuit emergency makeup in the pressure range of 8,0-0,1 MPa (80 - 1 bar), as well as a spray function.

The active emergency cooling system (cf. Figure 8) through the secondary circuit is made up of four closed circuits of secondary coolant cooling - one per each steam generator. Under normal operation these circuits are used for steam generator boiler water blow-down cooling.

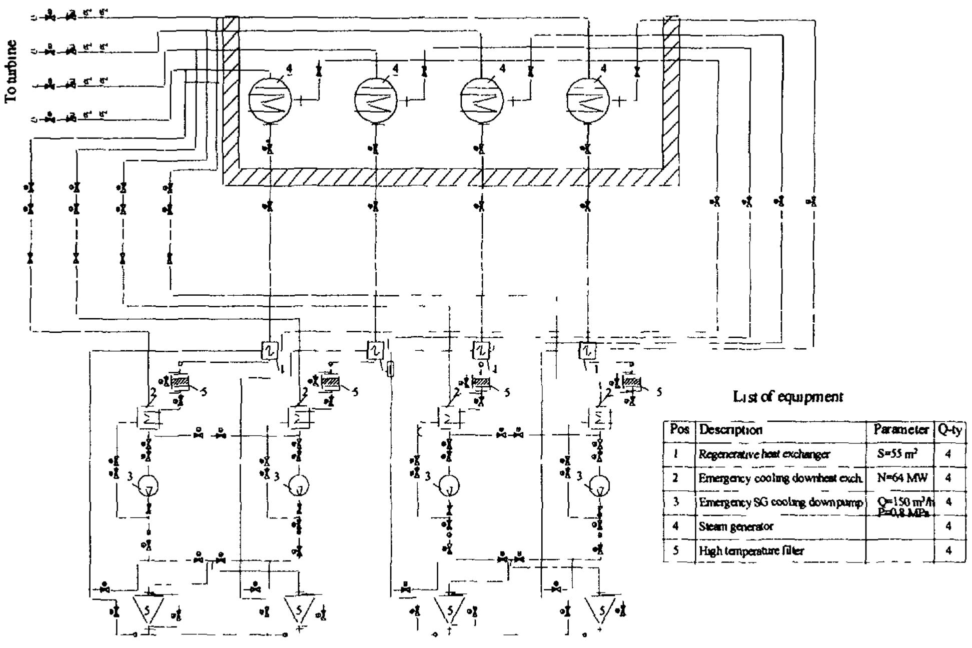
The passive quick boron supply system facilitates reactor shutdown in case of control rod system failure (conditions without scram). The system has four subsystems; each subsystem has a tank



Pos	Description	Parameter	Q-ty
1	Cooling down heatexchanger	F=935 m²	4
2	Cooling down pump	Q=230 m³/h	4
3	Water-jet pump (main)	Q=230-800 m³/h cold	4
4	Water-jet pump (auxiliary)	Q=20-30 m³/h cold	4

Note
For simplification only one channel with aux water-jet pump is shown

FIG. 7 Emergency cooling system for primary circuit and spent fuel pool



Pos	Description	Parameter	Q-ty
1	Regenerative heat exchanger	$S=55\ m^2$	4
2	Emergency cooling downheat exch.	N=64 MW	4
3	Emergency SG cooling down pump	$Q=150\ m^3/h$ P=0,8 MPa	4
4	Steam generator		4
5	High temperature filter		4

FIG. 8 SG blow-down and emergency cooling system

with concentrated boron solution connected to an RCP intake, discharged by pipelines, i.e., it represents a RCP bypass. If the solid absorber system fails in an event requiring the reactor to trip, the tank is connected to the loop. In this case, the boron solution goes to the primary circuit via the RCP intake. Inventory and concentration of the boron solution is selected to ensure compliance with safety criteria, in case of design-initiating events and where the reactor trip system fails to actuate. The system also functions where RCP power is lost, since the boron solution from the tanks is forced into the reactor due to RCP coast-down.

The passive system for heat removal from the primary circuit (PHRS) (cf. Figure 9) consists of four groups (corresponding to the number of SGs) of closed natural circulation circuits. In the ribbed tubular heat exchangers of these circuits, steam extracted from the steam generator condenses, and the condensate flows by gravity down the letdown pipelines into the steam generator boiler water volume. The PHRS heat exchangers are cooled by atmospheric air coming to the heat-exchanging surface near the draft air-duct outlet, through special direct action control gates that maintain steam generator pressure at a higher than nominal level. This solution makes it possible to prevent heat losses under normal operation and at the same time, keep the PHRS circuits warm. In case of station blackout, this solution prevents adverse primary circuit dynamics (coolant cooling, pressurizer level walk-away etc.). The cooling circuit is fully integrated and the operational duration of the PHRS is practically unlimited.

The passive system for reactor flooding (cf. Figure 10) during primary leaks comprises two groups of accumulators:

- 1st stage accumulators - 4 tanks of 50 m^3, each with a gaseous nitrogen cushion pressurized to 6,0 MPa (60 bar) and connected in pairs by pipelines equipped with check valves to the upper and lower reactor plena, through special nozzles in the reactor pressure vessel;
- 2nd stage accumulators are 8 tanks of 120 m^3 each, connected to the primary cold leg through check valves and to the primary hot leg through special spring-type valves. These valves are kept closed by primary circuit media pressure; when the pressure drops below 1,5 MPa (15 bar), the spring opens the valves. Such a connection configuration and valve design ensures continuity of hydrostatic pressure irrespective of primary pressure variation. Installation of throttling devices ensures a step-wise limitation of water drainage flow-rate with a decrease in

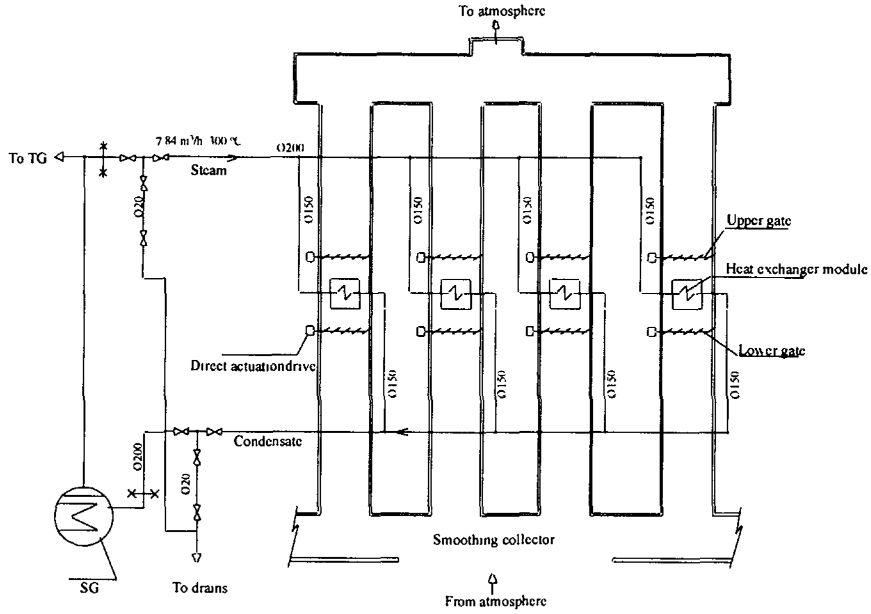
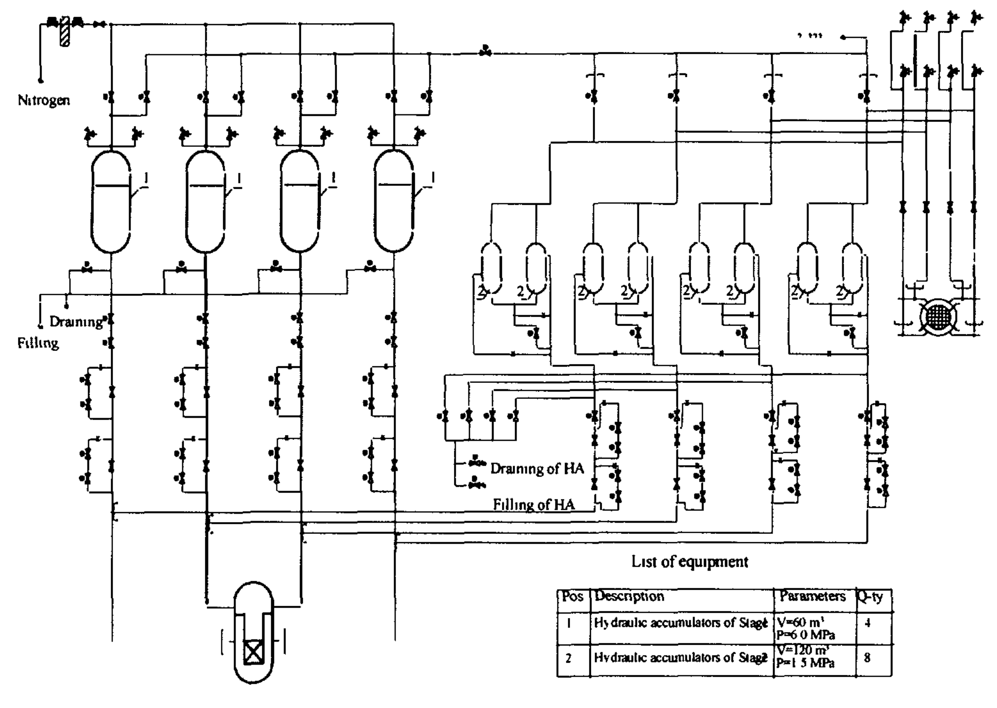


FIG. 9 Passive Heat Removal system



Pos	Description	Parameters	Q-ty
1	Hydraulic accumulators of Stage	V=60 m³ P=6 0 MPa	4
2	Hydraulic accumulators of Stage	V=120 m³ P=1 5 MPa	8

FIG. 10 The Hydraulic Accumulators system

the 2nd stage accumulator level. Stepwise limitation of the drainage flow-rate follows the law of decay power decrease with the necessary margin. The water inventory in the 2nd stage accumulators enables reactor cooling for 24 hours in case of leaks, even under blackout conditions, with all active mechanisms inoperable.

If AC power is not recovered after 24 hours, further core cooling is possible by PHRS. This system maintains low temperature in the steam generator boiler water and ensures the condensation of primary coolant steam inside the steam generator heat exchanging tubes, under conditions with the coolant level in the reactor pressure vessel below the hot nozzles. The evaporated coolant condenses, and returns, gravity-driven, along the loop pipe back into the reactor pressure vessel to cool the fuel.

3.3 Accident Product Confining System

The design basis sequence of loss of primary coolant accidents is overcome by the actuation of process systems (accumulators, emergency core cooling systems, containment spray system, containment isolation, etc.). Design values for the containment ambient under design basis accidents are:
P = 0,5 MPa; T= 150 ^{0}C.

Under the beyond design basis accidents considered in the design when fuel cooling is impaired, the limit accident performance values are established as follows:

P = 0,7 MPa; T = 200 ^{0}C.

The containment protection is provided by independent systems in the same way as for other critical safety functions:

- spray system (conventional);
- passive heat removal system; and
- an ultimate over-pressure relief device equipped with a high efficiency filtering plant.

The filtered discharge system is not actuated earlier than 12 hours after an accident initiation. To limit considerably the release of fission products beyond the containment, a permanent under-pressure is maintained in the inter-containment gap. This safety function, one of the most important, is fulfilled by two systems:

- an exhaust ventilation system equipped with a filtering plant - with suction from the inter-containment gap and outlet into the stack;
- a passive system of suction from the inter-containment gap. This system consists of lines connecting the inter-containment gap with the PHRS exhaust ducts, which are in a hot state. This solution enables permanent removal and purification of inner containment leaks irrespective of NPP power availability and operator actions. According to estimations, an under-pressure is maintained at any point in the inter-containment gap with inner containment leaks up to 2,8 per cent of volume per day (the design leak value is 0,3 per cent of the volume per day).

A hydrogen suppression system is designed to prevent hydrogen burning or explosion in the containment. The system comprises passive catalytic hydrogen igniters of efficient high porosity cellular materials. An element of 1,5 x 0,3 x 1,4 m dimensions oxidizes 30 l/h of hydrogen; its volumetric concentration is 4 per cent. Fifty igniters prevent explosive concentrations of hydrogen under beyond design basis accidents, when 100 per cent of Zr reacts with steam and hydrogen generated from other sources.

Table 2. Contributions to core damage frequency by different initiating event categories

Initiating event category	Contribution to core damage frequency (CDF)	
	absolute, 1/year	relative, %
1. Internal event during reactor power operation	2.6 E-8	48
2. Internal event at shutdown conditions	2.2 E-8	40
3. Seismic effects	5.9 E-9	11
4. Fires in the NPP premises	4.0 E-10	1
All event categories	5.4 E-8	100

For a beyond design basis accident leading to core damage and melting through the reactor pressure vessel, a core catcher is designed to make it possible to retain the corium within the compartments with a refractory coating. It is then cooled by passive means, with the help of accumulated water at the initial stage and by return of the condensate generated in PHRS heat exchangers.

3.4 Probabilistic Safety Assessment (PSA)

NVNPP-2 Unit 1 design includes performance of probabilistic safety analyses comprising:

- PSA-1 for internal initiating events;
- PSA-2 for internal initiating events;
- PSA for fires within the NPP premises; and
- PSA for seismic effects.

The main goals of PSA are to develop probabilistic models, assess probabilistic safety indices using those models and evaluate the safety level achieved based on the results. Core damage frequency and limit frequencies for large release of radioactive material to the environment were used as indices. Qualitative and quantitative evaluations of safety were performed. For the quantitative evaluation, the values established in OPB-88 (items 1.2.17 and 4.2.2) of 1.0 E-5 per reactor year for core damage frequency and 1.0 E-7 per reactor year for limit release frequency were used as goals.

The qualitative evaluation based on the PSA results was performed by assessing NVNPP-2 design compliance with the main engineering or deterministic safety principles, established in domestic regulatory documents (OPB-88 etc.) and IAEA materials (INSAG-3), which ensure the required level of defense-in-depth when met. Functional and structural diversity of safety systems provide deep protection against common-cause failures, and application of passive systems and active systems actuation without personnel interference yield deep protection against human error. Table 2 presents the estimated contributions to the core damage frequency (CDF) of different initiating events, including internal events probable during reactor power operation and shutdown, internal fires and seismic effects. The calculated total limit environmental release frequency over all the internal initiating events is about 4.77×10^{-8} 1/year.

Internal initiating events are the main contributors to the CDF (~88 per cent). The next contributor (~11 per cent) is seismic impact. Contribution from fires within the NPP rooms is relatively small (~ 1 per cent). So, the engineering solutions used in NVNPP-2 Unit 1 design enable attainment of a qualitatively new safety level, compared to its predecessors. The design meets all requirements of the defense-in-depth concept and target probabilistic safety indices established in regulatory documents.

Table 3 provides more detail with respect to different internal initiating events and Table 4 presents a comparison between the results for the NV NPP-2 and those for Unit 4 of Balakovo NPP.

4 EXPERIMENTAL AND ANALYTIC VALIDATION OF PASSIVE HEAT REMOVAL SYSTEM

Experiments and analyses were conducted to define PHRS operating parameters over the whole range of specified conditions with the aim of justifying design solutions for the passive heat removal system.

4.1 PHRS Experimental Validation

4.1.1 PHRS Experimental Study at Full-Scale Test Rig

Full-scale sections of the PHRS air heat exchanger-condenser with a design power of 5 MW were subject to experimental tests at OKB Gidropress test rig. The experiments were conducted with natural circulation steam condensate and air paths modeled; the environment air temperature was from - 19 ^{0}C up to + 30 ^{0}C and at steam condensing duct pressure from 0,5 MPa to 6,4 MPa. In the course of these experiments, the heat exchanger section thermal power was defined within the range of parameters specified during both operations with open gates and "waiting" conditions.

Table 3 Contribution to core damage frequency from different internal initiating events

Initiating event (IE)	Frequency, 1/year	Contribution to core damage frequency	
		absolute, 1/year	relative, %
1. LOCA into containment volume			
1.1. SLOCA	3,20 E-03	1,26 E-09	~2,6
1.2. MLOCA	1,00 E-03	3,64 E-10	<1
1.3. LLOCA	3,20 E-04	6,79 E-10	~1,4
2. LOCA to second circuit	1,00 E-03	1,26 E-09	~2,6
3.Reactor shutdown	1,00 E+00	7,38 E-09	~15
4. Loss of normal heat removal through secondary circuit	1,00 E-01	7,38 E-09	~15
5. Loss of offsite power	1,00 E-01	7,91 E-09	~16
6. Steam line break in part isolated from SG	1,00 E-03	2,67 E-11	<1
7. Steam line break in part not isolated from SG	4,00 E-04	1,29 E-10	<1
8. Loss of heat removal during shut-down and open pressure vessel	3,50 E-05	1,07 E-08	~22
9. Loss of offsite power during shutdown and open pressure vessel	3,70 E-03	1,12 E-08	~23
Total IE		4,77 E-08	100

4.1.2 PHRS Experimental Investigation at TDU-1 at IPE of Academy of Science of Belorussia

The experiments were performed using a double-loop and three-circuit installation of 1,0 MW. The installation had two identical loops with simulators of steam generators and air heat exchangers-condensers of 400 kW each. The availability of two loops made it possible to simulate the operation of two PHRS-SG circuits in parallel, under the conditions of heat exchangers non-equilibrium loading.

The experiments were performed with air temperature variations from + 5 ^{0}C to + 31 ^{0}C and steam pressure variation in the range from 0,6 MPa to 5,4 Mpa. The experiments demonstrated the stable operation of heat exchangers within the range of the parameters specified. No heat exchanging tube walls and condensate temperature variation was detected.

4.1.3 Experiments at "SPOT-2" Facility at IPE of Academy of Science of Belorussia

This facility simulates the main circulation circuit with VVER-1000 reactor and PHRS circuit in the power scale of 1:5500, maintaining the hydraulic similarity of full-scale and model circuits and actual differences in equipment location elevations. The experiments confirm the possibility of long-term
cooling down by the passive heat removal system for the case of an accident with main circulation pipeline rupture and loss of power. Under the conditions of experiment, the PHRS heat removal capacity was 97 per cent of the decay power.

4.1.4 Experiments for determination of wind effect upon RHRS capacity

When wind flows around the NPP main building, depending upon wind direction and velocity, a non-uniform pressure field is created along the containment circumference. It may cause air flow reversal in one or in a group of PHRS exhaust shafts. Atomenergoproject has suggested using common circular collector at the exhaust shaft inlets and one common collector with a deflector at the outlet to protect PHRS operation against wind effects.

An NPP main building simulator was developed and made in the scale of 1:80 to investigate the wind effect upon PHRS operation. This simulator was used to conduct experiments at Ts-22 NITs

Table 4. Comparison of CDF contributors for NV NPP-2 and operating unit 4 of Balakovo NPP

Initiating event (IE)	Frequency of IE	Contribution to core damage frequency			
		Unit 1 of NV NPP-2		Unit 4 of Balakovo	
		absolute, 1/y	relative, %	absolute, 1/y	relative, %
1. LOCA into containment volume					
1.1. SLOCA	3,20 E-03	1,26 E-09	~4,9	3,40 E-07	~0,8
1.2. MLOCA	1,00 E-03	3,64 E-10	~1,4	8,30 E-08	~0,2
1.3. LLOCA	3,20 E-04	6,79 E-10	~2,6	5,40 E-08	~0,1
2. Interfacing LOCA	1,00 E-03	1,26 E-09	~4,9	1,10 E-06	~2,6
3. Reactor shutdown	1,00 E+0	7,38 E-09	~28,6	1,65 E-06	~3,9
4. Loss of normal heat removal through the secondary circuit	1,00 E-01	7,38 E-09	~28,6	6,50 E-07	~1,5
5. Loss of offsite power	1,00 E-01	7,91 E-09	~30,6	3,54 E-05	~82,9
6. Steam line break in part isolated from SG	1,00 E-03	2,67 E-11	~0,1	3,40 E-06	~8,0
7. Steam line break in part not isolated from SG	4,00 E-04	1,29 E-10	~0,5	1,00 E-10	~0
Total IE		2,58 E-08	100	4,27 E-05	100

TsIAM in TsAGI. Aerodynamic tests on the main building simulator were carried out for the wind velocity range of 0 to 90 m/s (from calm to hurricane) with wind direction from 0 up to 360 degrees, in respect to the main building axis. The experiments performed on the NPP 1:80 simulator proved the correctness of design decisions to join the PHRS channels by a common inlet collector and common outlet collector, equipped with deflectors. No circulation reversal in the PHRS exhaust shafts was observed.

4.2 PHRS Parameters Analyses

Atomenergoproject carried out calculations to justify PHRS design parameters over the whole range of set conditions and under all PHRS modes of operation, including those under extreme temperature and wind effects. A number of computer codes were developed at Atomenergoproject, which facilitate calculations of PHRS steady-state and dynamic parameters, in particular, "RADUGA", "SPOT-KT", "GAMBIT", "STVORKA" codes. Using the above codes, calculations substantiating PHRS-related design solutions were performed and the effect of PHRS upon reactor plant operation was calculated for different accident conditions. Analyses of dynamic processes during PHRS operation with passive governor of air heat exchangers heat removing capacity were carried out. Analysis proving the optimum control device diagram was performed.

4.3 Passive Filtering System

4.3.1 Purpose of passive filtering system

The passive filtering system (PFS) is intended to control removal of the steam-gas mixture from the inter-containment gap under beyond design basis accidents with total loss of power. Prior to release of the steam-gas mixture into the atmosphere, it must be purified at the filters from radioactive substances entrained through the containment systems and elements entering through untightness into the inter-containment gap. The passive filtering system shall be operable under beyond the design basis accidents with loss of all AC power, both with tight primary circuit and with primary or secondary leaks.

4.3.1.1 System design

The passive filtering system can be divided into the following functional sections:

- inter-containment gap;
- stack;
- filtering device;
- air heater; and
- valves.

In a loss of coolant accident, the steam-gas mixture under high pressure and temperature, would appear in the containment. Through containment untightness (steel lining micro-cracks,. penetration untightness, cracks in concrete) the steam-gas mixture enters the inter-containment gap. Air heating in the stack at the expense of the pipe contact with hot air coming into the outlet collector from the PHRS heat exchangers, results in leveling the pressure differential in the passive filtering system. Due to this pressure differential the steam-gas mixture passes through the stack and is discharged into the atmosphere. Passing through the stack the mixture is heated, dehumidified and purified before release.

4.3.1.2 Mode of Operation

There are two modes of operation of the passive filtering system:

- waiting mode; and
- working mode.

Under normal Unit operation and design basis accidents the system is in the waiting mode. In the working mode, the passive filtering system should ensure rarefaction over the whole height of the inter-containment gap, as compared to atmospheric pressure.

4.3.1.3 System Operating Parameters

The greatest capacity of the passive filtering system in terms of clean air, provided rarefaction is maintained over the height of the inter-containment gap, is 0,066 kg/s for the worst external conditions. This value is equivalent to the total untightness of the containment, or 1,5 per cent of the containment volume per day.The containment pressure is 0,5 MPa(abs.), i.e., it exceeds design leak value by 5 times.

4.3.2 Experimental Study of Steam-Gas Media Flow through Concrete Wall Cracks

This experimental installation is designed and built with the purpose of conducting research into steam-gas propagation through cracks in the concrete wall. The results demonstrate that within the concrete block temperature range of 20,0 – 100 ^{0}C, there are no moisture drops at the concrete crack outlet. The absence of drops indicates that the filtering plant will not be subject to moistening. No drops were detected at the crack outlet during an increase in flow rate of pure steam as well as steam-air mixture at concrete block temperature of 20 ^{0}C. Block heating over 100 ^{0}C resulted in drops appearing from the concrete crack.

Additional design illustrations and safety data are provided in Figures 11 to 17.

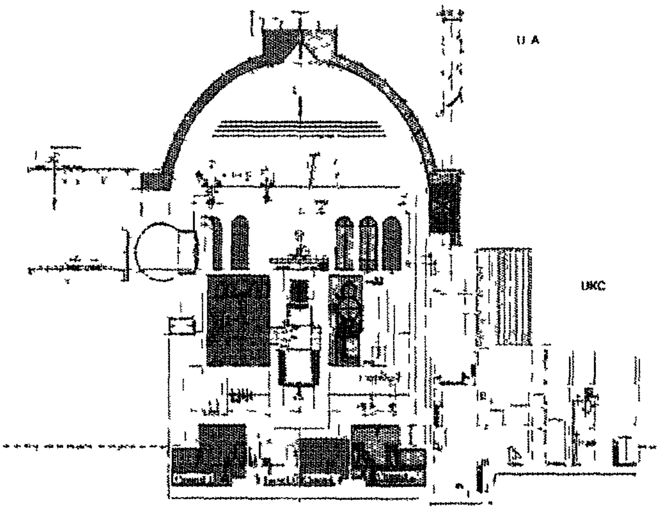


FIG 11 Section through Reactor building and Process building

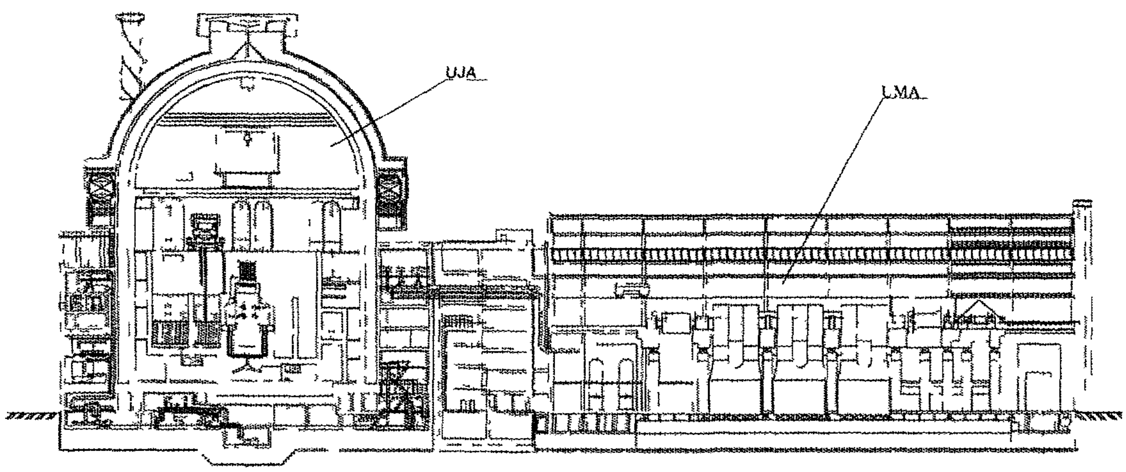


FIG. 12 Section through Reactor building and Turbine building

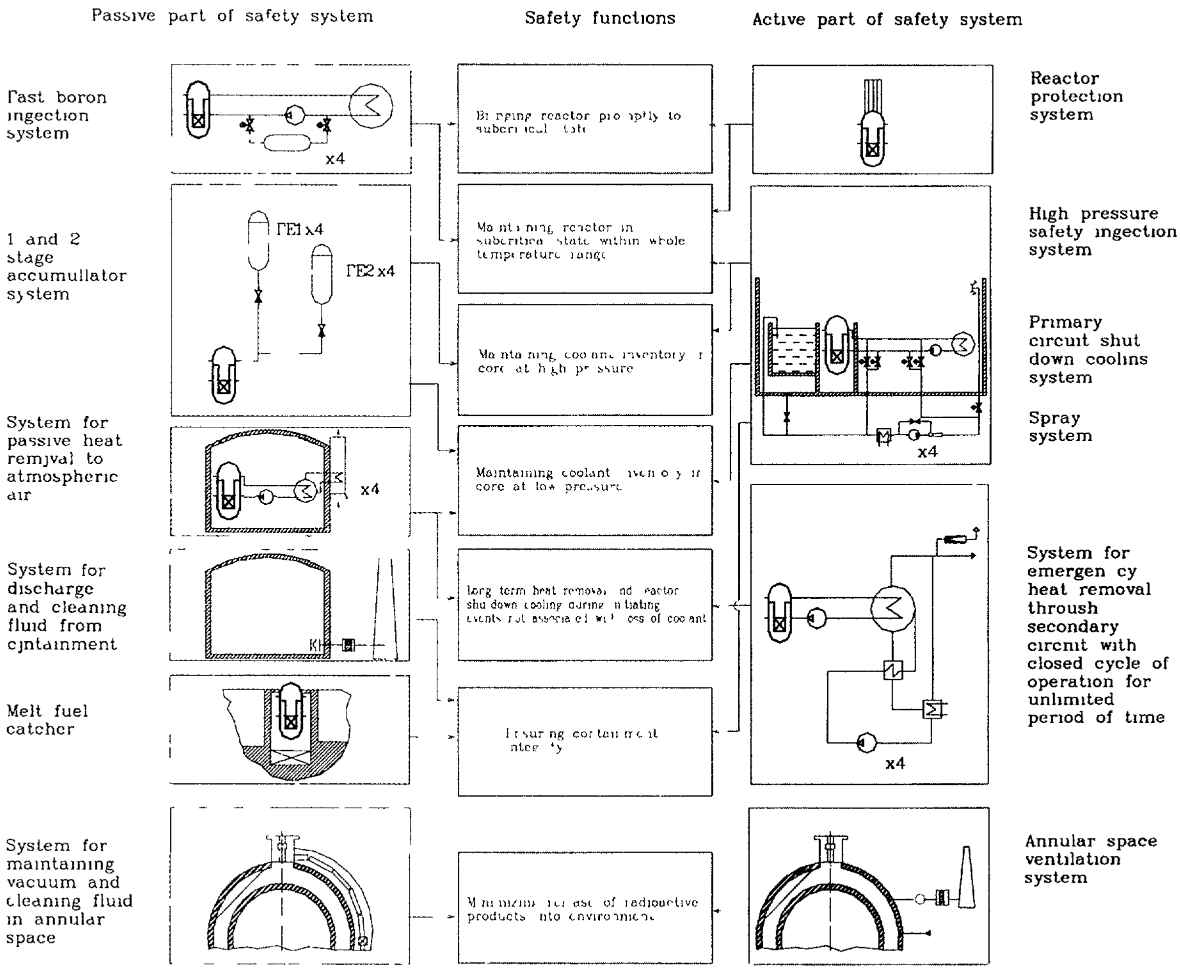


FIG. 13 Overview of principal approaches to ensure safety function in V-392

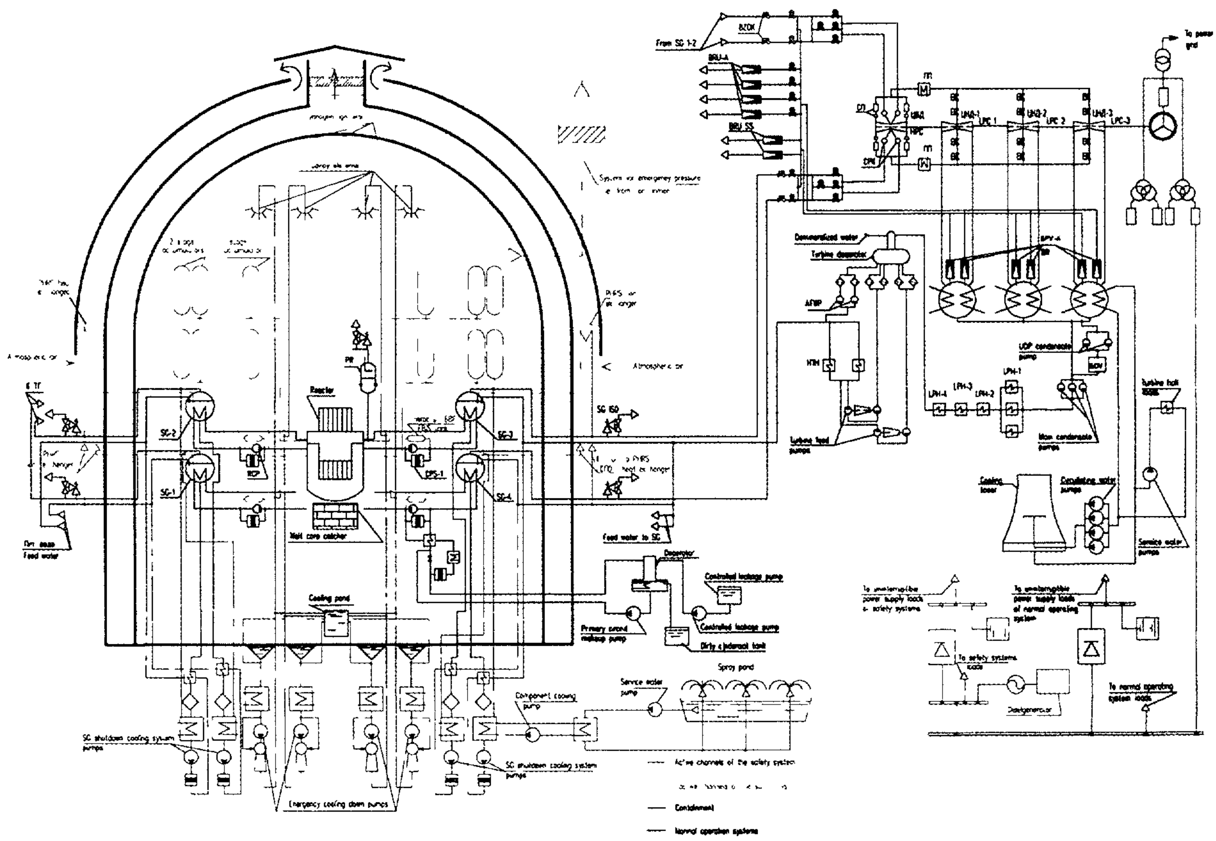


FIG 14 Basic diagram of NV NPP-2

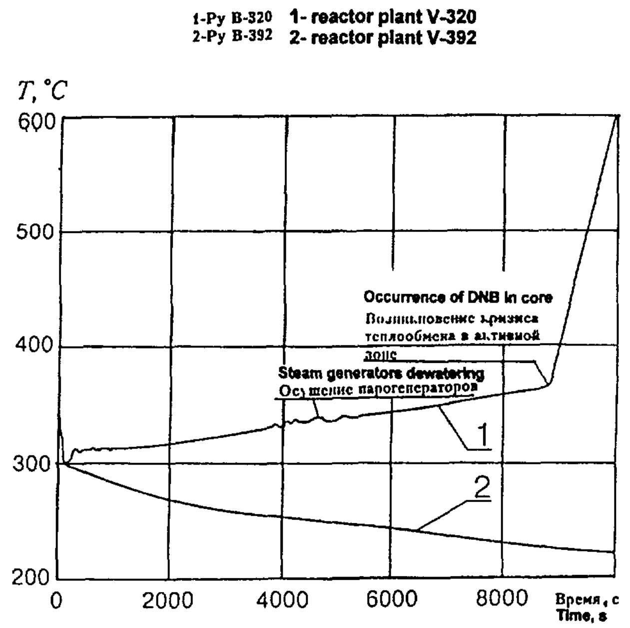


FIG 15 Max. fuel cladding temp. upon total loss of AC power

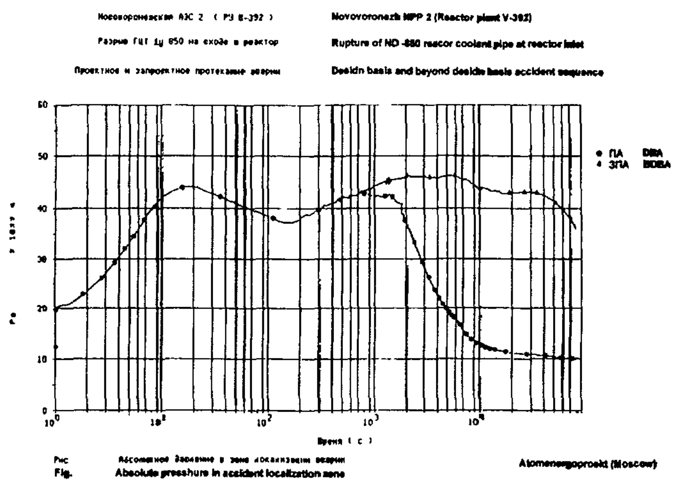


FIG 16 Rupture of cold leg pipe (DBA and BDBA)

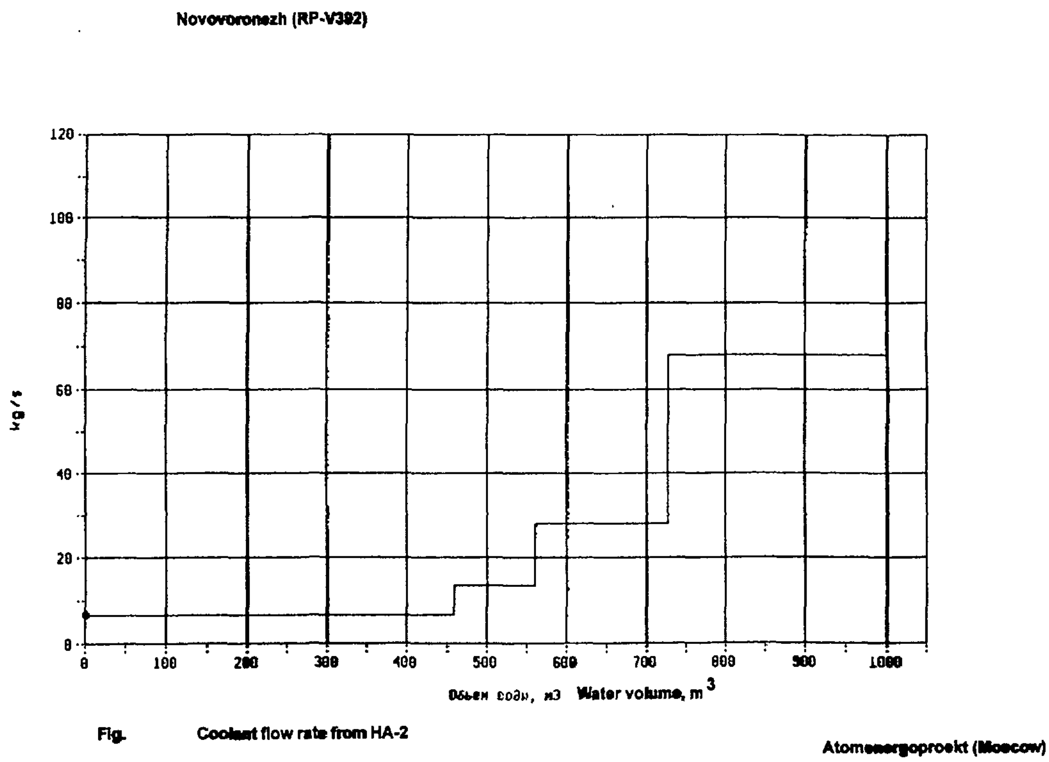


FIG. 17 Coolant flow rate from 2nd stage accumulators

POSTER SESSION I

IAEA-SM-353/1P

XA0053585

IMPROVEMENTS IN ARGENTINE OPERATING NPPs AND THEIR FUEL CYCLE

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The two power plants currently operating in Argentina are heavy water cooled and moderated, designed for natural uranium (NU) fuel and on-power refuelling. Atucha-1 (360 Mwe) is a Siemens design, pressure vessel type with vertical channels and full-length fuel elements, while Embalse (648Mwe) is a Candu-6 NPP with short bundles in horizontal channels.

The original fuel is natural uranium in both cases but for some time, use of slightly enriched uranium fuel (SEU) has been considered an interesting option, to significantly improve the exit burnup and correspondingly reduce fuelling costs.

In January 1995 the utility, Nucleoeléctrica Argentina S.A, initiated the SEU (0.85% enriched) fuel loading in Atucha-1 and in mid-year, about 100 of the 252 fuel channels were loaded with the new fuel. The irradiation program proceeded satisfactorily, and the goal of 11000 Mwd/ton has already been achieved as exit burnup of the enriched fuel. This is a remarkable increase over the previous 6000 Mwd/ton of the natural uranium core. This program is being conducted by the utility in close cooperation with CNEA. Only minor modifications of the fuel design were introduced and no plant hardware had to be modified. The PCI criteria were re-analyzed and translated to practical operating procedures for the in-core fuel management.

A series of studies and engineering activities were conducted in the frame of the enrichment program, related to the operating plants and with reference to the Atucha-2 plant, also of Siemens design, still under construction.

Plant start-up: SEU or NU fuel?

Many studies have been performed in Argentina on the use of slightly enriched uranium in the PHWR nuclear plants. These refer mainly to operating plants so a transition from the present natural uranium fuel cycle to the slightly enriched one had to be considered. Little has been said about the start-up core of a new plant.

In this analysis, technical and economical arguments are presented favouring the use of a natural uranium initial core. Technically, we can say that for a PHWR with on-power refuelling, the initial core is unique in the life of the plant. During the long start-up period, a very high excess reactivity must be controlled and this situation is never repeated. The plant is thus operated during a long period with high poison content, producing a reduced mean exit burnup of the first core. For the SEU case (0.85% enriched), the poisoned start-up core lasts a longer period because the fuel must achieve approximately twice the burnup of the NU core, before the excess reactivity reaches the standard equilibrium value. This reactivity excess is roughly the same for both cases because the operational transients are almost equivalent in this regard.

For licencing, the NU initial core solution is preferred because the reactor is much less reactive during the long transition from the start-up to equilibrium. Moreover, the economical side of the problem does not favour an initial SEU core. The levelized fuel costs are shown to be practically insensitive to the first core and a fast transition is more influential than an initially enriched core. In our case, where the sources of investment are scarce, the differential cost between a NU and SEU initial core is relevant. A more flexible fuel design plays an important role in speeding this transition and in reaching high load factors, of major importance in the economy of the plant.

Fuel element optimization

Step 1 :

An improved fuel element design for a PHWR using SEU fuel is studied. It maintains the general geometric disposition of the current fuel in the Argentine NPP's reactors, replacing the outer ring of rods with rods containing annular pellets. Power density reduction is achieved with modest burnup losses and the void volume in the pellets can be used to balance these opposing effects. The results show that this new fuel can be operated at higher power without violating thermo-hydraulic limits and this means an improvement in fuel management flexibility during in-core fuel shuffling as well as in the transition from NU to SEU cycle.

With respect to the void coefficient of the PHWR cell, it can be reduced and even reversed in sign by a combination of enrichment and some specific absorber distribution. It was demonstrated that using 2% enrichment in the outer ring of rods and a combination of depleted uranium with 1% Dy in the inner rods of the standard 37 rod Candu-type bundle, the same performance of the slightly enriched uranium (0.85% U235) can be achieved. The cell presents an interesting negative void coefficient but the internal power distribution is, of course, worsened with respect to the standard cell. This difficulty could be circumvented by using a zirconium liner. There is a margin for analyzing different solutions reconciling these two conflicting requirements: low void coefficient and a good power distribution in the bundle.

Spatial effects have been studied in this highly non-homogeneous cell, concluding that even with differences in the prediction of important parameters using a simplified or a more precise spatial model in the new cell, ring homogenization is reasonable and can be used for design.

Step 2:

A redesign of the PHWR fuel element CARA [3] has been proposed for all Argentine NPPs. The goal is to combine economical benefits with improved core-performance. The main features of this new bundle are an increased subdivision of the fuel material: 52 rods, and a 100 cm-long bundle. The neutronic aspects of CARA have been studied [4], showing that the in-core performance is similar to the standard fuel in Embalse reactor, whereas it improves considerably in Atucha-1, mainly due to the increase in uranium inventory. Of course, in both cases, linear power densities are significantly reduced. A variant of the design, with 48 rods instead of 52, is proposed. No burnup losses are involved in the change and costs are reduced.

These rod-dense clusters are also interesting when considering the void coefficient because the graded enrichment necessary to turn it negative no longer conflicts with the internal power peaking factor due precisely to the diminishing mean linear power density.

Core reactor calculations for 0.85% enrichment in Atucha-1 show that similar fuel management strategies can be used with the original CARA bundles, its variant of 48 rods, or the present standard fuel, and the exit burnup is 10% higher in the first two cases. Figure 1 shows a cross section of the Atucha - 1 core and Figure 2 a radial distribution of maximum channel powers in the cases studied. The natural uranium curve is shown as reference. All other cases are 0.85% enriched.

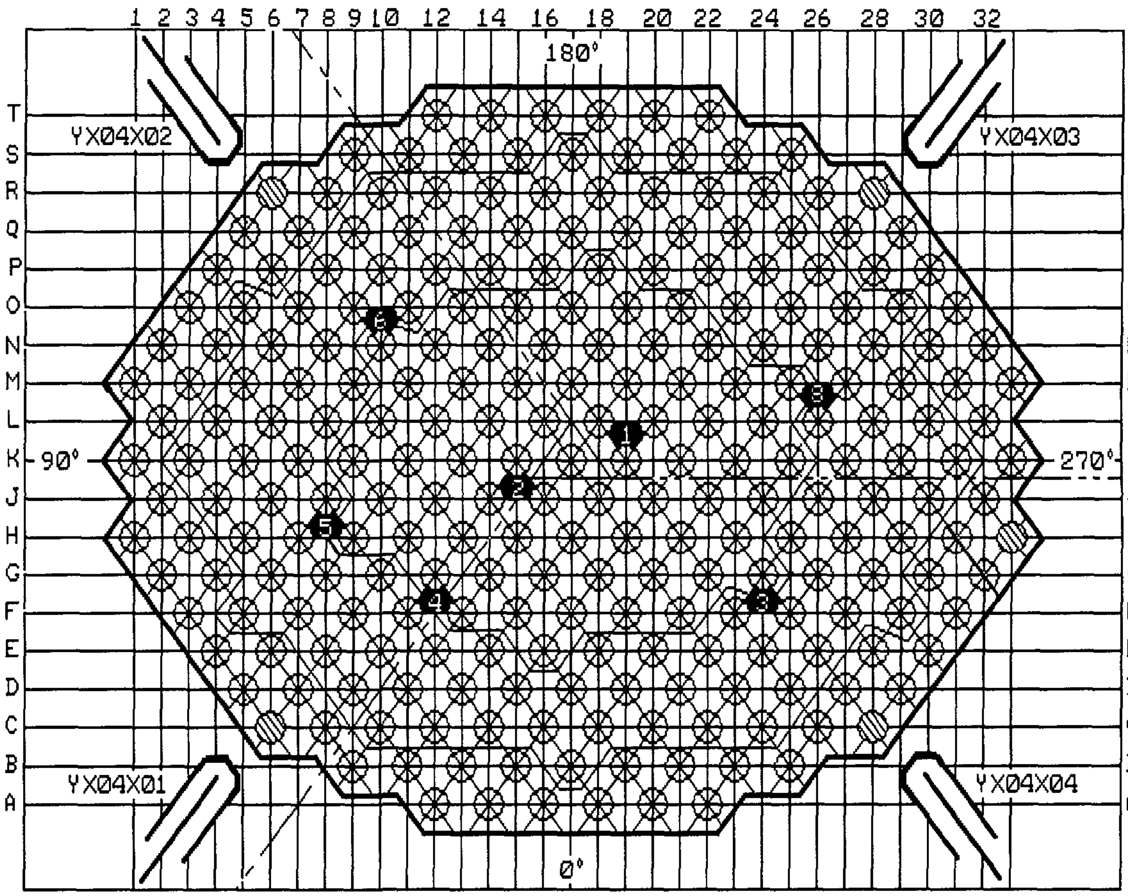

FIG. 1. A cross section of the Atucha - 1 core.

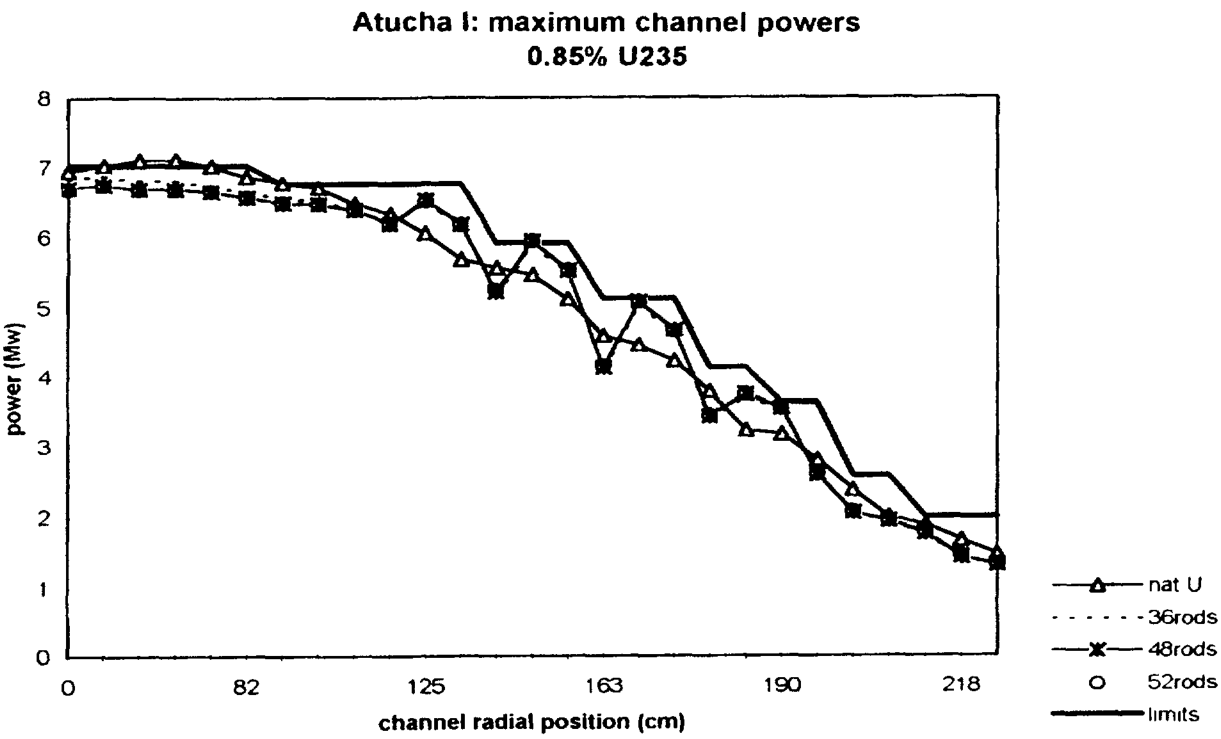

FIG. 2. A radial distribution of maxirnum channel powers.

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IAEA-SM-353/2P

THE NATURAL GAS NUCLEAR POWER PLANT: STRATEGIC PARTNERSHIP FOR ECONOMIC VIABILITY OF EVOLUTIONARY L/HWR

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Today, nuclear energy faces increasingly stiff competition with other energy sources. Where gas is available at low prices, combined cycle gas turbine (CCGT) plants appear especially formidable competitors against nuclear power plants. The relative cost composition of electric power differs in gas and nuclear generation. In the nuclear industry, operation and maintenance (O&M) and fuel costs are relatively low, while construction costs are the major component of investment. Contrarily, for the CCGT fuel costs are significantly higher than capital costs. Therefore, the economic competition between technologies is mostly determined by the relationship between the price of gas and the capital cost of nuclear plants, which varies from region to region.

Considering the most favorable scenarios, nuclear power costs at least 36 mills/KWh, whereas gas power generation ranges between 29 and 45 mills/GJ in Europe, and between 20 to 29 mills/GJ in Argentina, where the gas is abundant and inexpensive. Clearly, in regions where gas is available at low prices, generation by CCGT is a formidable competitor against nuclear energy. A major obstacle to profitability in nuclear power plants is the relatively low efficiency of their thermal conversion cycles. In effect, water-cooled reactor performance is limited by the primary coolant temperature, which cannot exceed 300 °C due to materials constraints.

During the 60's, when the thermal efficiency and reliability of LWR were still low, a few nuclear power plants with secondary overheating using fuel oil were constructed (i.e., Indian Point 1 in USA, Garigliano in Italy and Lingen in Germany). However, the performance of the combined cycle was questionable, due to low load factors and material failures. Now, the technology of thermal power plants, nuclear and conventional, is much more reliable (90% load factors). Consequently, it is reasonable to reconsider the feasibility of combined advanced cycles that produce vapor by means of nuclear power —taking advantage of the lower thermal costs— and superheating the secondary flow by the exhaust gases from gas turbines [1]. The concept combines the lower costs of nuclear fuels with the higher thermal efficiency of CCGT.

An assessment of the economy of combined nuclear-gas power plants was performed, viewing the merger as a convenient "strategic alliance" between both types of fuels, which offers electric power at lower costs. An interesting conclusion of this analysis is that gas price is the parameter determining the competition between the energy sources. Wherever the gas is expensive, nuclear power plants would be recommended whereas, in regions where the gas is available at low prices, CCGTs are preferred. In this scenario, the combined nuclear-gas cycle appears as the most convenient alternative in the range of moderate gas prices.

Figure 1 shows different countries in a map of competitive regions. The map plots gas prices and construction costs of nuclear plants in a two-dimensional plane, where the best alternative for power generation can be visualized by zones. Interestingly, most of the countries fall in the region where the combined nuclear-gas cycle is more competitive. The dashed line indicates the competitive boundary without considering the combined option, that is, separately comparing nuclear against gas. It can be seen that many countries are close to this boundary and countries like the UK and Argentina are constructing CCGT as the best option compared with nuclear alone. It is worth noting that the scenario may change completely if the combined nuclear-gas cycle is seriously considered a feasible power generator.

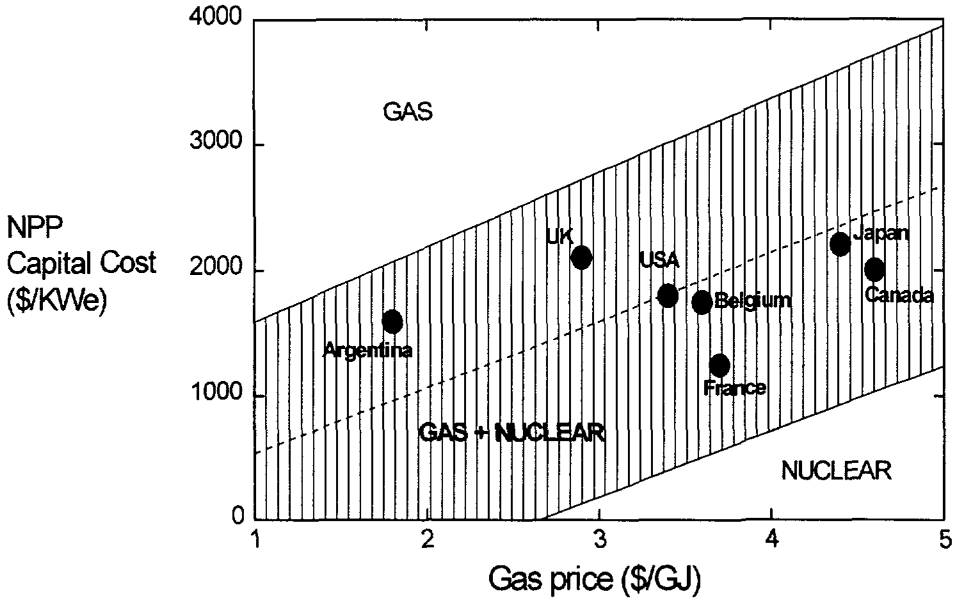

FIG. 2. Competitiveness map showing the most convenient alternative.

Contrary to trends from classical assessments of nuclear and gas power generation taken separately, the maximization of the superheated temperature was not found to be a good design criterion. In situations where gas prices are high, the optimum superheating temperature can be less than that technically achievable. Moreover, within rather wide cost ranges, the combination of nuclear and gas presents interesting possibilities for successful competition in the near future electric market.

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THE STUDY ON FEASIBILITY OF NPPs WITH EVOLUTIONARY WATER COOLED REACTORS FOR BELARUS

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The Belarus is in one of the lowest ranks with respect to the domestic supply of fuel and energy resources, even among the countries of Central and Eastern Europe. In 1990 it supplied a little bit more than 10% of its own energy needs. The annual cost of imported fuel and electrical energy (purchased mainly from Russia) at present prices in the CIS market, is about 1.5 bil USD, which exceeds the total state budget of the Republic. At world market prices, which we will reach in the next few years, this cost will increase by 1 bil USD. When energy consumption regains 1990 levels, the energy bill will reach 3.5 bil USD per year. Even now, one can envision chronic debt in the Republic of Belarus, to Russia for fuel and electric energy supplies; in the first half of 1998 it was not less than 200-250 mil USD.

Electricity generation facilities in Belarus include very old units. If this equipment is operated within existing rules, about 60% of it will be retired by 2005-2009. Most power units either have exhausted their operating life (300,000 hours for a turbine) or will be exhausted by 2005. Changes in the installed capacity of the Belarus electricity generating system, including retired units, is shown in Figure 1 (forecast of peak demand, the World Bank Report). The retirement schedule assumes that units are taken out of operation 300,000 hours after startup.

Scenarios for the electricity system expansion plan include only conventional technologies. Work in the Republic connected with preparations for NPP site surveys and construction are underway. The first stage of the siting process, according to the IAEA classification, is complete. It was based on criteria in a Safety Guide of the IAEA "Site Survey for Nuclear Power Plants" and requirements established in Russia for NPP siting. The results of preliminary studies show that there are at least three sites for construction. This allows including the nuclear power option in the list of possible technologies for electricity generation.

Using the WASP III Plus computer code, an optimal expansion plan was developed for the electricity generation system based on the installation of new combined cycle units and nuclear power plants. The optimal least-cost expansion plan was chosen from results of a comparative analysis of the three scenarios. Technology and fuel types considered for each case are summarized below: Technical and economical parameters are typical for technologies expected to be available during the planning period.

Characteristic	Scenario 1	Scenario 2	Scenario 3
Technology	Steam turbine, combined cycle, gas turbine	Steam turbine, combined cycle, gas turbine	Steam turbine, combined cycle, gas turbine
Fuel type	Natural gas	Natural gas, coal	Natural gas, Nuclear

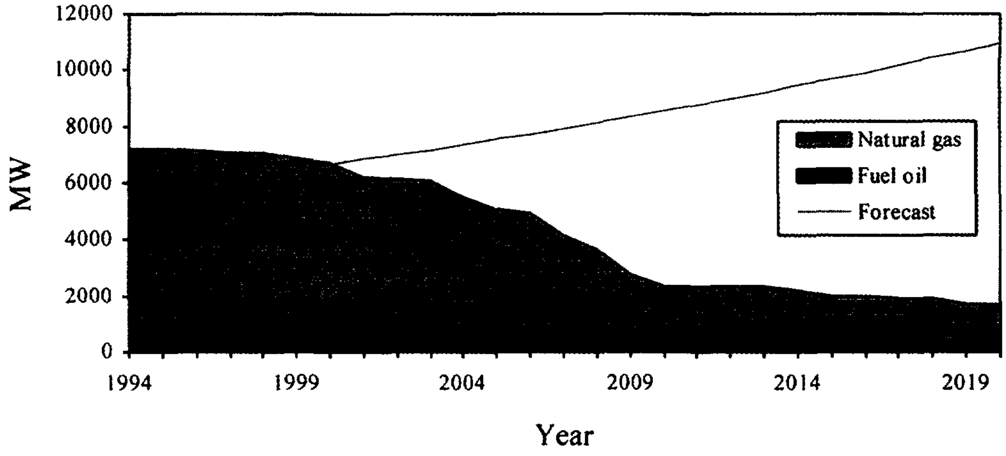

FIG. 1. Installed capacities of the Belarus electricity generation system.

A considerable growth of natural gas cost in 1998-2010 is connected, firstly, with a forecasted decrease in coal mining. The reason for this is exhaustion of the present deposits and the necessity to master extraction from new and almost inaccessible deposits. It is also necessary to consider the possibility of introducing a tax on burning organic fuels, to conform with world goals aimed at diminishing the danger of global warming. For Belarus, the forecast of prices for natural gas from Russia, at the moment and probably for the near future the main and possibly only supplier of natural gas, is of more significance. The analysis of fuel prices is summarized in Table 1.

Using WASP III Plus computer code, the optimal plan for the electricity generation system expansion was determined for each scenario. Calculations were made for a discount rate equal to 8 %. Electricity generation costs for the scenarios are shown below.

Scenario 1	3.60 cents/kWh
Scenario 2	3.62 cents/kWh
Scenario 3	3.26 cents/kWh

The results of calculations show that the electricity system expansion plan based on utilization of coal as a fuel has the highest generation cost. Contrarily, implementation of nuclear power will decrease generation costs up to 3.26 cents/kWh. In accordance with this, the optimal solution demands construction four nuclear units, with start-up of the first unit by 2010. Comparative analysis of the different scenarios is presented in Figure 2. Scenario 3, based on a natural gas and nuclear fuel mix, is expected to bear the least total cost electricity generation and is the most economically attractive option. Realization of Scenario 3 would allow a decrease in annual natural gas purchases and save up to 259 mil USD per year.

TABLE 1. FUEL PRICE FORECAST, $/GCAL

Year	Coal	Fuel oil	Nuclear	Natural gas
2000	12.3	9.6	2.33	11.7
2010	14.4	11.9	2.44	15.3
2020	16.8	14.8	2.57	19.1

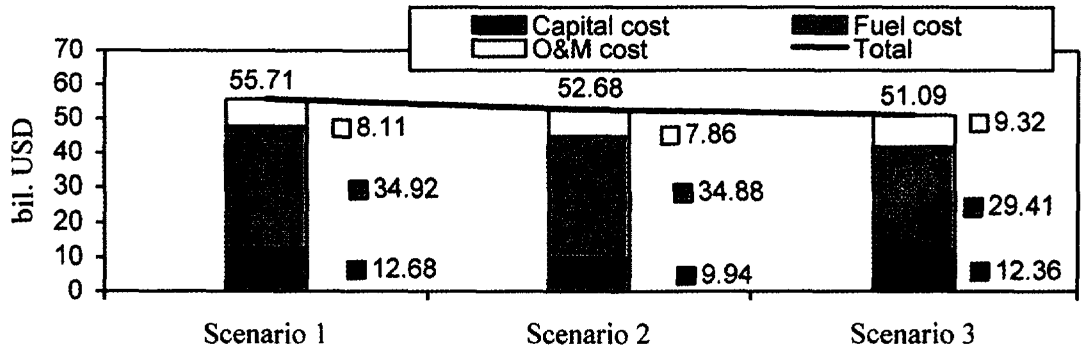

FIG. 2. Total cost of electricity generation during 25 years.

Annual capital cost for all scenarios is not very different. It means that, irrespective of the scenario chosen, the money required for the electricity system expansion is approximately the same. Thus, there is no reason to exclude the nuclear option based on the high cost of nuclear power plant construction.

The optimal structure of the electricity generation system by technology type depends on the historical hourly load and available fuel resources. Two peaks characterize the hourly load curve; the morning peak is at 10 am and the afternoon peak is at 6 pm. Both peaks are close, but the morning peak is slightly higher. To cover peak load demand, only 15-20% of the installed capacities were formerly needed. Most of the Belarus electricity generation system units operate in a base load regime. Therefore, the optimal structure of the technologies for electricity generation includes units traditionally considered most attractive for base load.

In accordance with results of the WASP runs, only 4-5% of the total capacity can be met by gas turbines. The optimal share of nuclear energy does not exceed 22% and approximately equals the share of steam turbines. As it has been calculated, combined cycle technology can provide about 55% of the required capacity.

IAEA-SM-353/5P

FUEL CYCLE FLEXIBILITY IN CANDU

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No single fuel-cycle path is appropriate for all countries. Many local and global factors will affect the best strategy for an individual country. Fuel-cycle flexibility is an important factor in an ever-changing, and unpredictable world. CANDU® is an evolutionary reactor, offering a custom fuel cycle to fit local requirements. Its unsurpassed fuel-cycle flexibility can accommodate the widest range of fuel-cycle options in existing CANDU stations.

In the short term, the CANFLEX® bundle is nearing power reactor implementation. The demonstration irradiation of 26 bundles in the Pt. Lepreau power station in New Brunswick, Canada, started this September, and a water-CHF test will take place later this year that will confirm the improvement in thermalhydraulic performance. CANFLEX provides enhanced operating and safety performance through a reduction in peak linear element ratings, and a significant increase in critical channel power. CANFLEX will be the near-term carrier of advanced fuel cycles.

While natural uranium fuel provides outstanding advantages, the use of slightly enriched uranium (SEU) in CANDU offers even lower fuel cycle costs and other potential benefits, such as uprating capability through flattening the radial channel power distribution. Recycled uranium (RU) from reprocessing spent PWR fuel is a subset of SEU that has significant economic promise. The use of SEU/RU is the first logical step from natural uranium in CANDU.

For countries having LWR reactors, the ability to use low-fissile material in CANDU reactors offers a range of unique synergistic fuel cycle opportunities. At one end of the spectrum is conventional reprocessing. The recycled uranium can be used directly in CANDU without re-enrichment. The plutonium can be utilized in conventional MOX fuel. Finally, the actinide waste can be mixed with some of the plutonium, and incinerated in an existing CANDU reactor using an inert-matrix carrier, such as SiC, burning most of the fissile plutonium, and a significant fraction of the actinide waste. At the other end of the spectrum is the DUPIC cycle, employing only dry thermal/mechanical processes to convert spent PWR fuel into CANDU fuel. With no purposeful separation of isotopes from the fuel, the DUPIC process has a high degree of proliferation resistance. This option is potentially simpler and more economical than conventional reprocessing. The first DUPIC elements have been fabricated by AECL and KAERI staff for a demonstration irradiation in the NRU reactor, to begin later this year. Between these two extremes of recycling options are a multitude of other advanced options that offer particular advantages in exploiting CANDU's high neutron economy to reuse spent LWR fuel without need of separating, and enriching the contained fissile material.

Thorium is a nuclear fuel in which there is potential widespread interest in the longer term, and short-term interest in those countries possessing extensive thorium resources, but lacking indigenous uranium supplies. Recycling in a CANDU reactor the ^{233}U produced by irradiating ThO_2 can reduce mined uranium requirements by up to 90%. Complete independence from uranium is theoretically possible with the self-sufficient-equilibrium thorium fuel cycle in a CANDU reactor, which in equilibrium, produces as much ^{233}U as is consumed. The full exploitation of the energy potential from

CANDU® and CANFLEX® are registered trademarks of Atomic Energy of Canada Limited (AECL) thorium requires recycling, which will not be economically justified for many years. Since commercial thorium fuel recycling facilities have not been built, there is an opportunity to develop a new, cheaper, proliferation-resistant technology for recycling.

In the long term, the CANDU reactor is synergistic with FBRs, with a few expensive FBRs supplying the fissile requirements of cheaper, high conversion-ratio CANDU reactors, operating on the thorium cycle. Direct recycle of fuel between the two reactor types is feasible.

The once-through thorium (OTT) cycle provides a bridge between current uranium-based fuel cycles, and a thorium fuel cycle based on recycle of ^{233}U. The optimal OTT cycle is economical today, in terms both of money and uranium resources. This cycle creates a mine of valuable ^{233}U, safeguarded in the spent fuel, for future recovery predicated by economic or resource considerations. AECL has recently devised practical OTT strategies.

The fuel cycle path chosen by a particular country will depend on many local and global factors. The CANDU reactor has the fuel cycle flexibility to enable any country to optimize its fuel cycle strategy to suit its own needs.

IAEA-SM-353/7P

MODERN METHODS IN CONSTRUCTION AND DESIGN FOR CANDU REACTORS

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There is increasing pressure to bring nuclear plants into production as soon as possible once the decision to build has been made. Many tools and techniques are available to the constructor to allow selection of an optimum combination of schedule and cost.

This paper deals with three major construction methods that are being utilized in the construction of CANDU products, and the tools that are facilitating the detailed planning that is required to meet the shortened schedule. These tools are also improving the quality of the engineered product to ensure construction is done right the first time.

CANDU 9 has been designed with ease and speed of construction as a requirement and modern tools and methods are being utilized to give confidence in an optimized schedule.

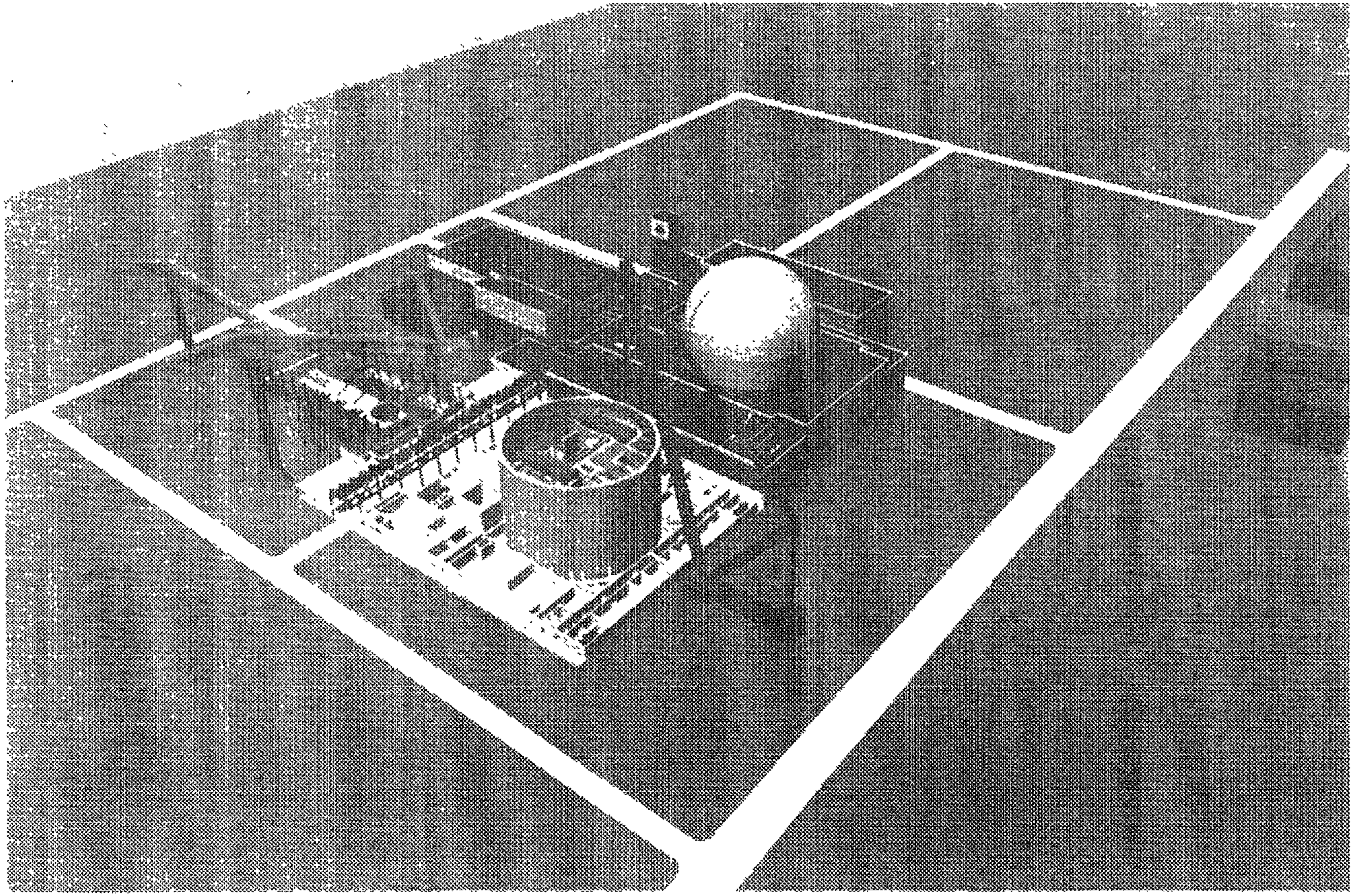
An overall construction strategy was developed leading to a target construction period of 56 months from the Limited Work Authorization to commercial operation for the first CANDU 9 unit in Korea.

- "Open Top Construction" means direct installation of most material and equipment into the Reactor Building utilizing external cranes prior to installation of the dome.
- "Prefabrication" can take many forms, from civil structures to mechanical/electrical skid mounted packages and may be of many different sizes.
- "Parallel Construction" techniques are those that redirect the sequence of events for Reactor Building construction from activities in series to activities in pararrel. With parallel construction techniques the mechanical construction program can be integrated into concurrent work areas with the civil program.

3D CADDs has been used by AECL for 10 years as a design tool. For the last few years this has been carried further utilizing advanced computer modeling systems which are used as both design tools, and for developing construction sequences and detailed planning. This allows the constructor to visualize multiple construction scenarios for the project in order to evaluate conflicts and risk. The development of PRIMAVERA schedules is an iterative process in combination with the planning sequences.

The paper specifically looks at an example of detailed planning, utilizing 3D CADD produced construction sequences. The example is for the prefabrication and installation of a major assembly, the Feeder Header/Pipe Whip modules. This module is assembled on site, in an area adjacent to the Reactor Building. This work is conducted off the critical path and it is not until the final installation, taking only a few days, that the work impacts the critical path. If this work had been constructed in the traditional piece by piece manner, the duration for these critical path activities would have been extended by several months increasing overall project duration. The Construction sequences allowed an iterative scheduling process to be conducted leading to optimization of the schedule.

The example clearly demonstrates that to ensure the success of a short construction schedule there must be very detailed construction planning. The project schedule will be 'construction driven' and, in particular, material deliveries must meet construction requirements. Well proven techniquies such as 'open top', prefabrication and parallel construction, combined with new technologies such as 3D CADD sequences allow detailed planning to increase confidence in meeting compressed schedules and reducing project risk.



Using 3D CADD to develop construction sequences and optimize the project schedule

XA0053590

IAEA-SM-353/11P

AVAILABILITY IMPROVEMENT PROCESS DURING THE DESIGN OF THE EPR PROJECT

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1. BACKGROUND OF THE CIDEM PROJECT

To minimize kWh costs, EDF decided to implement the CIDEM project (French acronym for Design Integrating Availability, Operating Experience and Maintenance), an analytic and systematic process for studying new projects, aimed at design optimization including investment, maintenance and availability objectives [1]. The first CIDEM application is focused on future French nuclear unit construction (the REP 2000 Program), within the framework of design studies of the new French-German Nuclear Island (European Pressurized Reactor [EPR]). This approach could be applied to other reactor-types or fossil-fired units specifically for its methods.

Directly based on EPRI/Utilities Requirements Document and European Utilities Requirements, a number of goals were set for the REP 2000/EPR project intended to maintain the overall costs of future generating units at a level competitive with other generating modes. These goals are:

- availability of at least 87%; and
- maintenance costs equal or less than those of current units.

2. FIRST STUDIES COMPLETED DURING THE BASIC DESIGN PHASE

CIDEM studies for the REP 2000 Program began in early 1993 and a number of results were obtained in the various fields covered by CIDEM, most significantly in the availability field. At the end of the Basic Design in August 97, studies conducted by the German Utilities and EDF tend to prove that technically, the initial availability objective of 87% was increased to 90% (for a fuel cycle length of 18 months).

3. POSSIBLE IMPROVEMENTS

Within the framework of the Basic Design Optimization Phase of the EPR Project, complementary studies were included in the CIDEM project after analysis of remaining margins. The main possible improvements identified are:

- to decrease the duration of outages and define a better position for these outages;
- to take advantage of the EPR design to perform on-line maintenance; and
- to propose a batch modifications policy.

The present allocation for forced outages set at 2% per year could be improved by reduction to 1.4% (as required by EUR and URD). Larger studies are needed on Structures Systems and Components, with a possible impact on investment, but these studies are not presently planned.

3.1 Duration and positioning of the outages

Two different kinds of outages are needed to operate a nuclear power plant:

- the normal outages, which can include:
  ⇒ **NRO**, or Normal Refueling Outages, for core unloading-reloading and most preventive maintenance;
  ⇒ **ROO**, or Refueling-Only Outages, only core unloading-reloading;

- the long outages, which can include:
  ⇒ **TGO**, or Turbine-Generator Overhaul, Normal Refueling activities plus general inspection of the turbine and the main generator,
  ⇒ **ISI**, or In-Service Inspection (8-year inspection in Germany, 10-year inspection in France), which includes Turbine-Generator outage activities and most of the regulatory inspections (vessel inspection, RCS hydrotest, containment pressure test).

3.1.1 Normal refueling outages

The Normal Refueling Outage schedule was studied in detail with the German Utilities during the EPR Basic Design phase. Related requirements were sent to NPI and the design options not yet defined for the EPR project, allow attainment of the initial objective of a normal refueling outage in 19 days (from breaker to breaker). Shutdown and startup phases were optimized but to maintain some margins, it is not now possible to reduce their duration.

On the other hand, the duration of the work phase, including steam generator inspection and safety train maintenance can be reduced, for two main reasons:

- some provisions exist in the EPR layout; they allow use of robots for the SG inspection. To inspect 20% of the tube bundles during the outage (French practice), allocation of 92 hrs (168 in the previous schedule) seems realistic if the 4 SGs are controlled in parallel.
- the 4-safety train design allows performance of most of the preventive maintenance with the unit in operation. Work not feasible during plant operation would be performed during normal refueling outages. For each safety train, an allocation of 60 hrs is proposed.

Considering these new assumptions, the theoretical duration of a normal refueling outage is reduced to 382 hrs (16 days), compared to some German Konvoi outages performed in 15 days.

3.1.2 Refueling-Only outages

The "Refueling-Only" outages are not yet used systematically by EDF, but the first was recently implemented alternately with Normal Refueling Outages, into the outage policy for current plants. However, considering both maintenance and inspection constraints (especially regulatory), this outage policy can only be used for fuel cycles no longer than 12 months.

For the EPR project, the duration of a Refueling-Only Outage was determined from the Normal Refueling Outage schedule, by suppressing all maintenance. This achieved a theoretical duration of 275 hrs (~12 days). However, this result is determined by the ability to perform RCCA reshuffling in less than 8 hrs on the critical path. This assumption must be confirmed.

3.1.3 T/G Overhaul

Two strategies can be considered:

- either group all the maintenance activities on the T/G into a single long outage; or

- split the overhaul into several Normal Refueling Outages (HP cylinders, LP cylinders, Generator).

Considering the batch modifications policy proposed in §2.2 and the duration of Refueling-Only Outages (12 days) and Normal Refueling Outages (16 days), it is impossible in practice, to perform the maintenance, for example of HP cylinders, during a short outage. Therefore, CIDEM proposes grouping all maintenance on the Turbine-Generator during long outages (TGO and ISI). Under this assumption, the duration of a TG Outage would be 31 days.

3.1.4 In-Service Inspection

The previous assessment of the In-Service Inspection duration for the EPR was performed and was the same duration as the shutdown, the core unloading and reloading and the startup phases rather than that for the Normal Refueling Outage. The duration of regulatory inspections came directly from the reference schedule of current 1300 MW units (EDF reference UTO 93). The resulting duration was 49 days.

Analysis of the last updated schedule (EDF reference UTO 96) of the 10-year In-Service Inspection for current units shows a significant reduction in the duration of regulatory inspections; the total outage decreased from 93 to 68 days.

Considering these new data, and an increased pressure for the containment pressure test (6 bars instead of 4), the new assessment of the In-Service Inspection duration for the EPR is 312 hrs (38 days).

3.1.5 Positioning of the outages

The figure in the Appendix shows positioning of the outages for different fuel cycle lengths, on a reference period of 10 years, to account for all kinds of outages. This is summarized below:

Number of outages	**Fuel Cycle Length**		
	12 months	**18 months**	**22 months**
Refueling Only Outage	**4**	**0**	**0**
Normal Refueling Outage	**3**	**4**	**3**
Turbine-Generator Overhaul	**1**	**1**	**1**
In-Service Inspection	**1**	**1**	**1**

3.2 Outage extensions for modifications and miscellaneous works: Proposal of a policy for modifications

Analysis of the impact of modifications on outage duration in current plants shows two kinds:

- simple modifications; and
- batch modifications.

For the EPR, CIDEM proposes to perform simple modifications during NRO; these modifications induce outage extensions for work no longer than 9 days. The batch modifications, more complex and weighty, are made during long outages, TGO and ISI. The extensions for work are set at 19 days for ISI and 9 days for TGO. For the ROO, work of 3 days is proposed.

3.3 Outage extensions for unplanned events

The realization of modifications and miscellaneous work induce extensions from unplanned events no longer than :

- 5 days for ROO and NRO,
- 10 days for TGO and ISI.

3.4 Other unavailabilities

The assumptions concerning other unavailabilities are the same as those fixed during previous studies:

- yearly forced outages - 175 hrs (i.e 2%);
- yearly unavailability for tests - 20 hrs; and
- power increasing (from breaker to 100% NP) - 24 hrs.

4. CONCLUSIONS

Considering these results, the availability values to retain for cost assessments are :

- 90.2% for a 12 months fuel cycle;
- 91.4% for a 18 months fuel cycle; and
- 92.2% for a 22 months fuel cycle.

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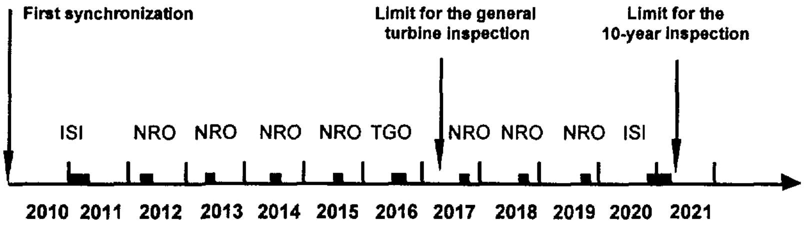
APPENDIX

EPR PROJECT - POSITIONING OF THE PLANNED OUTAGES

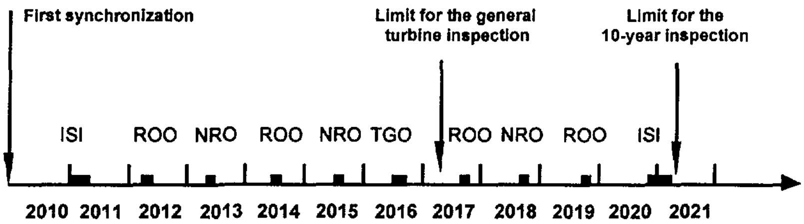
7 Normal Refueling Outages

12 months cycle



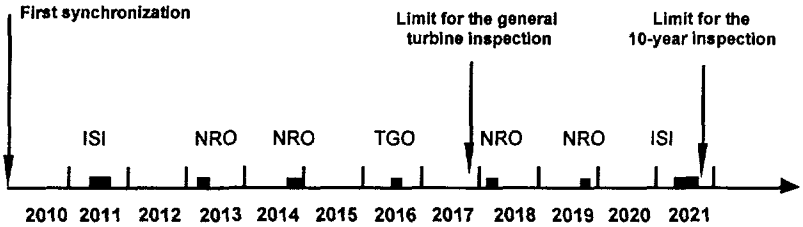
4 Refueling-Only Outages
3 Normal Refueling Outages

12 months cycle with ROO



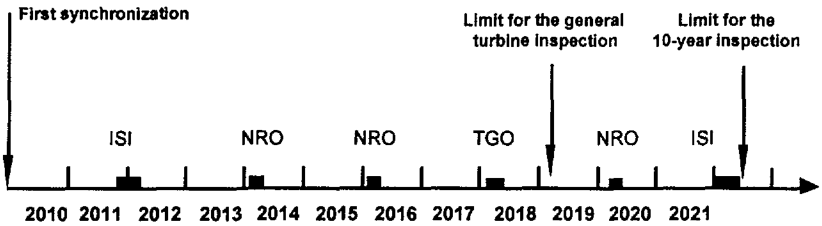
4 Normal Refueling Outages

18 months cycle



3 normal refueling outages

22 months cycle



NRO : Normal Refueling Outage
ROO : Refueling-Only Outage
TGO : Turbine-Generator Overhaul
ISI : In-Service Inspection

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IAEA-SM-353/13P

SAFETY-RELATED INNOVATIVE NUCLEAR REACTOR TECHNOLOGY ELEMENTS R&D — SINTER NETWORK

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As in any other applied technique, nuclear reactor technology still possesses an important potential for innovation to respond to the steady evolution of requirements with regard to safety, economics, waste reduction, operational characteristics, complementary applications, etc. This potential must be included in judgements on the 'Nuclear Option' within sustainable future energy supply scenarios and social environments.

The technical improvement process itself is characterized by research & development (R&D) concerning a multitude of technologies as well as scientific and technical disciplines covering the complexity of the nuclear island and related infrastructure. Each of these "Technology Elements" has its own developmental scope and time frame that may sometimes collide with the tough schedules and budget limitations of commercial projects, although they may open new ways of coping with existing or arising challenges. Future reactor concepts will use these innovations by combining them with proven technologies and systems, creating the base for innovative reactor concepts that respond to the evolution of public and industrial requirements flexibly and effectively.

A sustainable R&D strategy for nuclear reactor technologies should therefore involve a balanced coexistence of short, medium and long-term R&D goals, as is the case in other commercial or competing areas. Technological evolution always comprises a range of steps as well as the feedback of experience to effectively explore the viability of new technical "mutations".

Key actions towards a consistent and sustainable R&D strategy, replacing or interconnecting national programs, are required on a European level, to ensure and shape the future of nuclear power as an active reaction to the evolution of public and industrial requirements [1&2]. With this background, the SINTER Partners proposed ways the R&D on innovative nuclear reactor technologies could be incorporated into ongoing programs in Europe and ways this field interacts with R&D on existing reactors and commercial projects. The result of this discussion was to characterize the area of R&D in nuclear reactor technologies in three levels with regard to the different objects, time horizons and main players:

1. Level: R&D on Existing Reactors
2. Level: R&D for Commercial Projects
3. Level: R&D on Innovative Nuclear Reactor Technologies

The *third level* incorporates the *pre-competitive medium- to long-term R&D aspects* that may open new technological options for future nuclear reactors with improved safety characteristics, for waste reduction or for improved and economic operation. In this sense, this R&D level is complementary and interconnected to the previous one and does not interfere with commercial activities. On the contrary, for long-term sustainable use of nuclear power it is essential to demonstrate that nuclear technology is not restricted to today's technical status but is capable of keeping pace with the general evolution of technologies and requirements, even in the long run. Innovative technologies may also help reconcile nuclear energy with public and political opinions, offering new arguments for those countries not yet opting for nuclear power.

The objective of this CEC-funded 'Concerted Action' is to conduct a survey of R&D primarily on safety-related innovative technology for future nuclear reactors, components and systems at a European level, in response to the evolution of public and industrial requirements. Modern telecommunication and telecollaboration techniques will be assessed and used to establish a database and new information channels to stimulate a broader cooperative network. The SINTER Network is closely linked to the industrial MICHELANGELO Initiative [2] which also aims to define a European R&D strategy towards improved safety and economics for the next generation of nuclear reactors.

The work program of the 'SINTER Network' relies mainly on available information on on-going and/or planned national, international and industrial R&D programs and will generate bottom-up input based on a questionnaire. Potential R&D Items will follow a classification system covering the main innovation areas of safety improvement, waste and cost reduction within the following work packages:

1. Assessment of Innovative Reactor Concepts and Future Requirements.
2. R&D on Innovative Safety Features.

3. Improvements of Fuel Characteristics and Efficiency.
4. New Reactor Designs with Emphasis on the Reduction of Waste and Exposure.
5. Enhanced Safety by New Technologies, Methods and Materials.
6. Conclusions and Database.

Another questionnaire collects information on medium to long-term R&D needs of 'end-users' such as supplier industries, utilities, licensing and governmental bodies in a top-down approach, as a complement to short-term industrial R&D.

In this context, a workshop on 'Risk perception as initiator and steering instrument of innovative R&D' was held on April 1-3, 1998, in Ascona, Switzerland. It addressed the question of how the perception of technical risks triggered and still triggers or influences technological developments and corresponding R&D. The results show that the evolution and mechanisms of social awareness do not accept a purely probabilistic view but need an active technological response, for example, in limiting *maximum releases, minimizing the production of radiotoxic wastes and reducing the timeframe for* potential dangers from waste disposal. The workshop came to the conclusion that nuclear energy must comply with the general criteria of sustainability to be an "acceptable" candidate for future energy supply systems. *'Mental'* and *'environmental'* acceptance together with economic competitiveness is seen as a precondition for broader commercial use of nuclear energy.

A registration procedure and an interactive questionnaire is now available via Internet ***http://www-is.ike.uni-stuttgart.de/sinter*** for up-to-date and flexible data acquisition and handling within the European nuclear community. Further information services will be offered to contributors and implemented step-by-step during the project, depending on response of the users.

It is possible to use the same methods in other R&D areas and programs to harmonize and exchanging selected data in the frame of future developments of nuclear reactor technologies on a broader international basis.

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STRATEGIC ISSUES IN EVOLUTIONARY WATER COOLED REACTOR: ASSESSMENT IN INDONESIA

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Increasing energy supply is a necessary condition that must be fulfilled in supporting national development to improve its quality of life and standard of living. Until the 1998 monetary crisis, Indonesia had a rate of electrical energy demand of more than 15% per year. For a developing country like Indonesia, with low gross national product per capita and limited energy resources, this condition poses serious problems in the long run. For that reason, as an integral part of our Long Term National Development Plan, it is necessary for Indonesia to include nuclear energy as an option for energy diversification. This is in accordance with national energy policy, which stipulates among other things: environmental concerns, diversification and conservation supporting sustainable development [1].

To justify the introduction of the first nuclear power plant (NPP), a comprehensive and deep feasibility study had already been done. One of the main results of the study was that nuclear energy would contribute to the primary energy supply scenario in the year 2004. However, there are still some strategic issues related to development of NPP in the area of technology, economic and politics. Those three issues are interrelated to each other leading to safety-related problems.

In technology, safety of the NPP is a special concern; that stems mainly from severe accidents, some safety related problems, and radioactive waste and environment. For a country like Indonesia, just beginning to introduce the NPP program, those three area are important in providing convincing proof to the public about the safety of NPP, leading to public acceptance. In a severe accident, especially one with major core degradation, not only should the probability of an accident decrease to a very small number, but also should the accident happen, the radioactivity must be contained such that no serious consequences are expected for the environment in the plant neighbourhood [2].

The safety aspect of the NPP is very fragile issue in light of public resistance. The Indonesian public is sensitive to the possibility of accidents in nuclear power stations, because for them, nuclear energy is something new. Thus any safety-related problem, either nuclear or non-nuclear should be suppressed to a small probability. As a consequence, the design organization must show that the safety performance of the nuclear power station is at a very high level. The last issue concerning technology is radioactive waste disposal. The radwaste disposal issue in Indonesia is clearly related to the site availability and location especially given that it is an archipelago of a volcanic nature.

In the area of economics, major issues are the cost of remedial action in case of an accident, and liability due to an accident. The 1997 Indonesian nuclear energy act (Act No. 10/97), includes some articles (Art. 28-40) related to liability in nuclear accidents [3]. In case of a nuclear accident, no other person than the operator shall be liable to provide compensation for the damage suffered by the third party. The maximum amount of liability is Rp.900 billion per incident. This new act concerning the liability will influence the economics of nuclear power plants, and in the end, influence the price of the energy produced.

Balancing the issues in the area of technology and economics is difficult; on one hand there should be a good and sound safety performance, sometimes costly, but on the other hand, still economically competitive compared to other energy sources. Those opposing forces result in the third issue, which is political in nature, that is the difficulty of decision making, especially with current monetary problems.

Having experienced these several weighty issues in the process of developing its first NPP, however, BATAN still endorses the development of the evolutionary water-cooled reactor with the requirement of meeting the technological and economical issues. This is because, up to now, the water cooled reactor is still the leading reactor with the longest operating experiences without significant failure in an active system. However an additional requirement still applies; that is, providing convincing proof that even in case of the most severe accident, accident management could detain radioactive releases inside the containment and prevent serious consequences to the environment.

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IAEA-SM-353/15P

APPLICATION OF NUCLEAR POWER PLANTS FOR HEATING IN KAZAKHSTAN

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One feature of the power resources in Kazakstan is that a major share, 55-60 per cent, is expended for heating in industry, social applications and housing every day. This is caused by the natural climatic conditions of the country. At present, power sources in the larger cities are combined heat power plants (CHPP) and boiler plants. Sixty to seventy per cent use coal as the basic fuel; almost all the rest use black oil, with only a few natural gas-operated. The large amount of low quality fuel burned every year results in major releases of hazardous substances, and, undoubtedly, impacts negatively on the environment in populated localities supplied with heat. Typically, small CHPP and boiler plants produce tens and hundreds of MW and are the basis for the heat-generating power park in the Republic of Kazakstan.

Under the existing structure and with the construction backlog, large coal stocks will dominate in heating, power plants and boiler plants to generate electricity and heat in Kazakstan for the next 40 to 50 years. Under current conditions, power engineering development and operation in the next ten years must solve many problems but most of all, environmental problems.

The situation is complicated because very often, heating sources are in cities and towns economically based on mineral, metallurgical and chemical industries and characterized by major energy use and large hazardous substances releases into the atmosphere. Such a combination creates difficult environmental situations in many cities and industrial centers because there is a great shortage of heat in these populated localities.

Specifically, the cities for which IAE NNC RK evaluated the installation of nuclear power for heating are Almaty and Leninogorsk. The Institute of Atomic Energy was also involved in developing materials for "The validation of investments for constructing a pilot-industrial demonstration atomic plant of low power (PID AP LP)", with construction expected on "Baikal-1" site of IAE NNC RK.

The computations and analysis confirm the validity of the proposals. So, in Almaty, the construction of a nuclear heat-generating plant of four units with reactor facilities of the type NHGP-500 in combination with the application of Balkhash APP, will supply the city with power. It will allow decommissioning Almaty CHPP-1, operating with old equipment using coal as a basic fuel and contributing the most environmental pollution, among power enterprises, to the city. To develop the options for nuclear power plant application, the following factors were considered:

1. Fuel flow rate, its reduction and prices for existing sources were specified based on data from POE&E "Almatyenergo" Report.
2. First of all, power-generating equipment with their current sources should not be located in the district-heating cogeneration zone because the environmental situation is complex and many industrial enterprises are too close to residential areas in the city's center. It is especially important because these basic power-sources in the district-cogeneration zone, particularly CHPP-1, operate with old equipment.
3. The specified heat and electric power of the nuclear sources should be not less than that of the power generating equipment it replaces.
4. The existing technological vapor sources should not be changed by nuclear power sources.
5. Alternative nuclear sources should be located so as to require minimum addition to the heat networks when they are connected into the present heating system.
6. The heating systems should be developed and improved within the scope and terms specified in previously prepared development plans for the city heating system.

Major parameters of the proposed options for nuclear power source application in Almaty heating system are presented in Table I.

From these data it appears that the most acceptable options are 3 and 5. These options provide the greatest saving of fuel and the greatest reduction of hazardous substance releases in comparison with the others. In addition, when realizing options 4 and 5 and, respectively, and in rejecting the construction of the South-Kazakstan HPP, a yearly expenditure for 16,25 ml.t of coal (93,19 ml. US$ for the price of 1991) is eliminated. Whereas, the price of construction-only for basic industrial HPP objects is 2 475,72 ml. US$ (in 1991 prices), and its specified power is 3240 MW, it is less than APP power, according to options 4 and 5, by 570 and 1160 MW respectively.

Option 5 is the most attractive . If realized, the CHPP-1 operating with old equipment and located in the center of the city and part of the equipment of the Western local boiler plant, Almaty HPP and CHPP-2 would be decommissioned. The construction of Almaty HPP-3 is not

TABLE I. MAJOR INDICATIONS OF ALTERNATIVE OPTIONS FOR UBSTITUTION OF THE EXISTING POWER SOURCES BY NUCLEAR ONES

	Option 1	Option 2	Option 3	Option 4	Option 5
Number of atomic plant unit	4×AST-500	4×ATET-200	4×ATET-200	AST500 6xNP500	4×AST500 4xNP1000
Specified nuclear Source power heat, Gcal/hr electric, MW	2064 0	1500 720	1500 720	2064 3810	2064 4400
Substituted power Heat, Gcal/hour Electric, MW	1800 0	1500 318	1255 655	1955 828	1955 828
Additional power, directly in Almaty heat, Gcal/hour electric, MW	264 0	0 402	95 245	109 0	109 0
Construction price, ml. $US	760	1 100	1 100	760+4 950	760+5 700
Fuel saving. Thous. t.sp.f.	471	748	999,4	1208,4	1208,4
Price of the saved fuel, ml. Tenge	487,45	594,9	640,57	813,3	813,3
Reduction of hazardous substances release, thous. T	10,203	19,267	23,356	28,546	28,546

needed, and there is a possibility of rejecting a part of the fuel imported from Central Asia. We would then be capable of exporting a large portion of the power generated due to a surplus in comparison with SK HPP power of 1160 MW. The construction cost of nuclear sources according to this option is 6 460 ml. US$; the time for realization of the proposed option is 8 to 10 years.

For nuclear power units close to Almaty, draft proposals have been developed applying current safety concepts, providing high operational safety, with a probability of severe accidents not exceeding 10^{-7} reactor/year. Under normal operations, the total radiation impact from the plant to the population and environment would not exceed 0.1% from the phone values .

Since the Almaty region is a seismic area (9-10 MSK number), construction of nuclear plants with new generation reactors of high safety type "AHP", "AHP" and "NG-500" /4-11/, which can operate in areas of high seismicity, is expected. Increases in the nuclear unit seismic resistance up to 10 in number is possible through application of low-frequency seismically insulated devices, reducing seismic loading by 10-30 times.

Based on the estimates, construction of a small dual-purpose, two-unit nuclear plant with reactor KTL-40 facilities is proposed for heat and power generation for Leninogorsk city and the Multimetal Plant. The basic advantage of the proposed reactor is positive experience in operational safety as power for atomic ice-breaking ships. It should be emphasized that nuclear reactor operation as ship power is characterized by highly complex and intense RF operation modes and high requirements in reliability and safety. KLT-40 reactors operating for hundreds of reactor-years on ships confirm that they fully meet the requirements with high reliability and operational safety.

The results from "Validation of the investments for constructing AHPP LP in Leninogorsk-City" illustrate that in spite of large capital expenditures, construction of this plant is economically effective. According to preliminary estimates the design NPV becomes positive and varies from some tens to 700 ml. US$, if the charge for electric power is within 5-15 cents/kWh, and the AHPP LP payback time does not exceed 25 years, in a given 50-year lifetime.

We consider the construction and operation of these power installations and a small experimental-industrial nuclear demonstration plant on the "Baikal-1" site complex of IAE NNC RK to be justified by the efficiency, ecological cleanness, safety and complete acceptance for nuclear facilities as local heat sources.

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DEVELOPMENT OF AN EVOLUTIONARY PASSIVE PWR CONCEPT: CP-1300

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DESIGN DIRECTION OF CP-1300

According to the government's current plan, Korea is to construct PWR and CANDU-PHWR power plants until the time when fast reactors are introduced. The PWRs include the 1,000 MWe KSNP (Korean Standard Nuclear Power Plant) until about 2010, and the 1300 MWe KNGR (Korean Next Generation Reactor) after 2010. Considering the rapid progress of technology, it is thought that a third-generation reactor, a passive PWR or an improved version of the KNGR, will be needed in late 2010s.

The Center for Advanced Reactor Research (CARR) is investigating the feasibility of a large passive PWR concept, CP-1300, to fill this role. The KNGR and KSNP are used as reference designs for the CP-1300, to take maximum advantage of domestic technologies. The CP-1300 plant will have a concrete containment and the final safety functions will be achieved through passive systems. The main features distinguishing the CP-1300 include:

- an increased number of fuel assemblies to reduce the power density;
- new core make-up tanks (CMTs) for smooth safety injection;
- decay heat removal by steam generator secondary side natural circulation; and
- passive containment cooling with internal or external condensers.

Since the CP-1300 program was initiated in 1992, the reactor concept was established, some preliminary safety analyses completed, and small-scale verification experiments for key design features are underway. As a result, the conceptual design and the preliminary design parameters of the plant have been determined.

OVERALL CONCEPT

The NSSS design of the KNGR (similar to that of System 80+) was adopted with some modifications for enhancement of safety and operational margins. The passive safety systems are derived mainly from Westinghouse's AP600 and GE's SBWR (now superseded by the ESBWR), with appropriate modifications. A double concrete containment system was adopted in consideration of safety, construction experience, and costs.

The reactor coolant system (RCS) consists of the reactor, two steam generators, four reactor coolant pumps, a pressurizer, two hot legs and four cold legs. Instead of the canned motor pumps used in AP600, centrifugal sealed pumps having large inertia were adopted. The sealed pump has the characteristic of slow flow coast-down compared with the canned motor pump.

The CP-1300 adopts the KNGR-type reactor core. However, the number of fuel assemblies is increased from 241 to 249 to reduce the power density. Therefore, core thermal parameters, such as the average heat flux on fuel rods, the average core power density, etc., are generally reduced and the operational margin increased.

To effect a passive residual heat removal system (PRHRS), "two-phase natural circulation of the steam generator secondary water through secondary condensers (SCs) submerged in a condensing pool located outside the containment" was adopted. It shows outstanding capability for depressurizing the primary system for loss-of-feedwater accidents, according to the preliminary safety analysis. Compared with other types of PRHRS, it provides a simpler arrangement of components inside the containment, thereby facilitating maintenance work with a smaller possibility of radioactivity release during a steam generator tube rupture accident.

To effect a passive safety injection system in the CP-1300, an automatic depressurization system similar to that of the AP600 provides for effective injection of IRWST water into the reactor at low pressure. Two direct vessel injection (DVI) lines are used for safety injection. Two core makeup tanks (CMTs) with spargers are used for high-pressure injection. To solve the boron dilution problem, the cold leg pressure-balancing line is eliminated and to assure an adequate driving force, spargers are attached to the pressurizer balancing lines in the upper part of the CMT. For long-term containment cooling, the internal condenser concept utilizing the natural circulation of the passive containment cooling system (PCCS) pool water through the condenser tubes inside containment was adopted as the basic option. As an alternative to the internal condenser concept, an external condenser concept, similar to the isolation condenser system in the SBWR, is also being investigated.

FUTURE WORK

In the future, a more detailed design will be developed and a comprehensive safety analysis, probabilistic safety assessment and economic assessment will be performed, along with separate validation tests on the new systems.

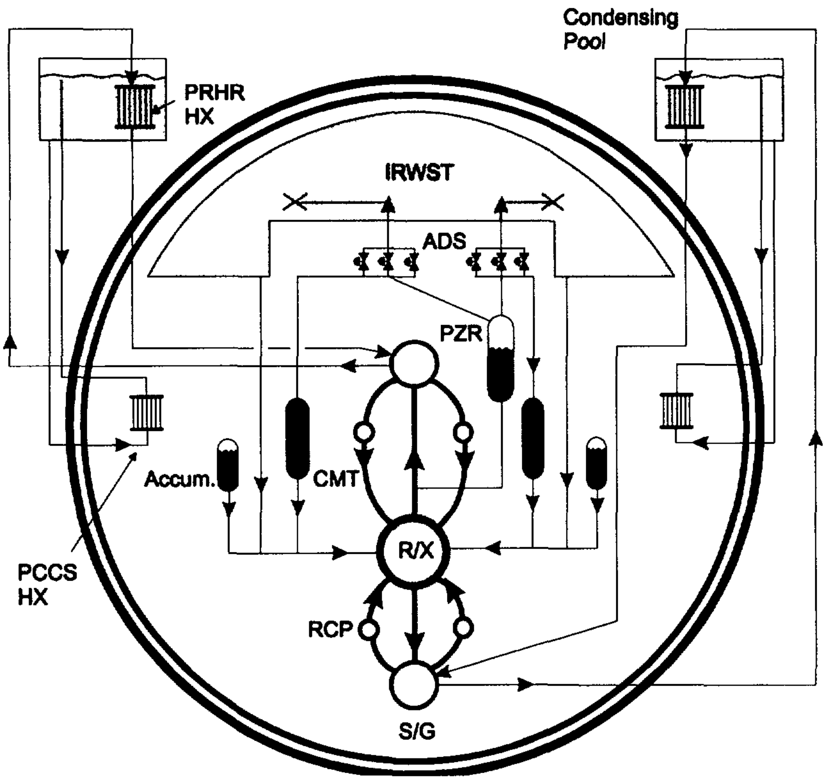

FIG. 1. Schematic of the CP-1300 safety systems.

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IAEA-SM-353/22P

DEVELOPMENT OF NEW LICENSING SYSTEMS FOR EVOLUTIONARY WATER COOLED REACTORS IN THE REPUBLIC OF KOREA

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1. Introduction

Under a comprehensive program to construct nuclear power plants, over the last two decades Korea has carried out and gained valuable knowledge on utilization and regulation technology related to nuclear power. Based on these experiences, Korea developed a regulatory infrastructure consisting of a licensing system, regulatory requirements, and necessary procedures.

Several Korean Standard Type Nuclear Power Plants have already been constructed. Recently, both evolutionary water-cooled reactors and advanced reactors have been under development in accordance with the National Long-term Development Program for nuclear plants. In consideration of these long-term goals, the Korean Government, Ministry of Science and Technology (MOST), and Korea Institute of Nuclear Safety (KINS) recognize the necessity of improving licensing systems to minimize complexity and uncertainty in future reactor regulations.

With the aim of developing a new system, current licensing based on a two-step approach such as Construction Permit (CP) and Operational License (OL) has been reviewed and re-assessed to enhance the effectiveness and efficiency of safety regulations. The licensing system under development consists of three processes: *Pre-application Safety Review, Standard Design Approval and Improved Combined Licensing for construction and operation.* These will be incorporated into the existing system as complementary measures for consideration of reactors such as the Korean Next Generation Reactor (KNGR), the Korean optimized Standard Nuclear Power Plant (KSNP), Liquid Metal Reactor (LMR), Small and Medium Reactor (SMR), etc.

2. Pre-application Safety Review

As early as possible in the development stage of evolutionary water-cooled reactors, the utility wishes to secure the licensibility of proposed designs with advanced features such as passive safety systems, digitalized instrumentation and control systems, etc., so development can proceed in a stable environment. In this regard, the Government, MOST and KINS introduced the Pre-application Safety Review (PSR) to nuclear industries, to encourage advanced interaction of applicants with the regulatory body, not only for early identification of regulatory requirements but to provide more timely and effective regulation. In addition, the PSR system enables the reflected comments of related parties, including the public, from the view point of the regulatory body, concerning the desired characteristics of advanced reactors. Such interaction and guidance early in the design stage contributes toward minimizing complexity and enhancing stability in the licensing.

In the PSR, existing regulations, and Codes and Standards for conventional nuclear power plants are basically applied first to advanced reactors. However, when addressing regulatory requirements for application to new characteristics of design, additional regulatory criteria will be developed and applied to future reactors thereafter. Consequently, it is expected that the pre-application safety review process would be conducted prior to the formal licensing process of a future advanced reactor.

3. Standard Design Approval

The standardization of nuclear power plant designs is important for the utility because it significantly enhances safety, reliability, availability, and economy. Approved standard designs benefit public health and safety by concentrating resources on specific design approaches, by stimulating standardized construction practices and quality assurance, and by fostering more effective maintenance and operation. Accordingly, it is expected that standardization of nuclear plants could further improve safety in future plants and promote more efficient review. Approval of a standard design could be attained prior to construction in accordance with an improved licensing process to be introduced later.

The objectives of the standard design approval system are twofold as follows:

- Encourage the use of standard plant designs to enhance plant safety and improve the effectiveness of the regulatory process.
- Improve the efficiency of licensing review by early resolution of safety issues for the timely construction of nuclear power plants.

4. Improved Licensing Process

The Korean regulatory system currently employs a two-step licensing approach based on the prescriptive regulation, similar to practices in the U.S. However, future plants, such as the KNGR, will be designed through the concept of standardization so as to continuously construct succeeding units. Introduction of a new licensing system including Standard Design Approval aforementioned, and an Improved Combined License, applicable to standardized plants is currently being considered in Korea. This is a modified and improved system developed from the existing two-step licensing to establish a one-step license for construction and operation, including a fuel loading permit.

Much care, however, should be taken in the introduction of such standardized licensing because it may have an adverse impact on the Korean regulatory environment. In-depth evaluation of the merits and drawbacks will be performed before its introduction. When introduced, it is expected that the new system will be applied to the licensing of future nuclear power plants.

5. Future Prospects

The Government, MOST and KINS believe the new licensing systems, namely *Pre-application Safety Review, Standard Design Approval, and improved combine licensing,* will enhance nuclear safety, and encourage and promote further standardization. These three systems are under review, possibly to be incorporated into the soon to be finalized Atomic Energy Law.

These new licensing systems would give licensees greater assurance to develop standard designs and contribute to increased stability and predictability of regulation for future nuclear power plants. In addition, when the new licensing system is a reality, it will benefit both the regulatory body and utility in terms of time and resources, enabling both sides to exert increased effort toward assuring safety while simultaneously enhancing efficiency.

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JOINT DESIGN DEVELOPMENT OF THE KNGR MAN–MACHINE INTERFACE

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The Korean Next Generation Reactor (KNGR) program is developing a standard Advanced Light Water Reactor (ALWR) design for operation after the year 2007. This cooperative development effort between the Korean nuclear design organizations has completed the Planning and Conceptual Design phase and is currently in the Basic Design phase. The KNGR design is an evolutionary advancement from the current generation Korean Standard Nuclear Plant (KSNP) design. Some of the most evident changes are in the KNGR Man-Machine Interface Systems (M-MIS), where state-of-the-art technologies are replacing conventional control rooms and Instrumentation and Control (I&C) systems.

The KNGR MMIS distinguishes itself from current designs by employing all digital I&C systems and data communications and primarily video-based Man-Machine Interfaces (MMI) which incorporating modern human factors principles. The MMI maintains proven KSNP features, such as critical function monitoring, while adding advanced design features addressing the Electric Power Research Institute's ALWR Utility Requirements Document. Advanced design features being incorporated include: (1) a compact workstation-type control room layout, (2) a large display panel, (3) computerized operating procedures, (4) soft controls, (5) a reduced number of fixed location displays and alarms.

A structured MMI design process is being coordinated with the overall KNGR Program design phases. In late 1997, a Joint Design Development team convened at the initiation of the Phase II Basic Design effort. An experienced team of Korean and ABB-CE control complex designers and I&C engineers was assembled, complemented by experienced PWR operators. The charter of the Joint Design Development team was to determine features to be incorporated in the MMI conceptual design. The team spent three months in an intensive evaluation of proposed MMI design features with respect to their impact on operability, the I&C system designs and licensing. Of particular importance was maintaining the existing licensing basis established for an advanced control complex of a U.S. ALWR design. The Joint Design Development resulted in selection of conceptual design features to serve as a point of departure for the KNGR control room layout, man-machine interfaces and I&C systems, including updated design and licensing documentation.

The KNGR MMI design effort is continuing in 1998 with formal documentation and design details being developed for each MMI feature. Systematic evaluations of the design features are also being conducted. Dynamic prototypes of all design features are being developed for initial evaluation. Actual man-machine interfaces are then being developed for a selected set of plant systems. These will be incorporated into a dynamic mockup, representing a major part of the control room, to allow operator-in-the-loop evaluations. The mockup is being driven by plant simulator models, allowing continued evaluation of the individual design features through suitability verification. It will also provide an opportunity for initial evaluation of the KNGR MMI ensemble through pre-validation transient exercises.

The results of the Phase II will support formal submittal of the MMI licensing documentation by the end of the year and provide high confidence reference designs for beginning the Phase III Detailed Design. In parallel with design efforts, the formal human factors design process has also begun. Initial activities include operating experience review and a functional requirement analysis and functional allocation. This process will continue throughout the year.

The primary control and monitoring facilities for KNGR control room operators are full-function compact workstations, which provide unprecedented operational flexibility and integration. Each of three redundant workstations features paired sets of monitoring CRTs and soft-control flat panels, allowing access to, and the effect of, both safety and non-safety controls. Complimenting these soft interface devices, computerized operating procedures reduce operator response time variability and operator error probabilities. This full complement of soft devices accommodates varied task requirements and operator preferences, and permits incremental MMI refinements with minimal impact throughout the plant design process and operating life. A Large Display Panel is located in the front of the control room. It provides a fixed-location indication of high priority alarms, parameters and component status otherwise unavailable in the compact workstation environment. It also provides continuous display of critical safety function status, per regulatory requirements. A minimum set of fixed-position component controls and operator modules are located on a safety console, designed to complement the workstations during post-trip conditions. The KNGR control facilities support control room operator staff reductions, compared to conventional designs, while offering unprecedented expansion and reconfiguration potential.

This paper discusses the results of the KNGR MMIS Joint Design Development effort, including the MMI conceptual design features. It also describes the continuing Phase II design and evaluation. Brief descriptions of the design features, evaluation process and evaluation results are provided. The paper concludes with a look to future KNGR MMI development and licensing activities as Korea's next generation control room takes shape.

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THE DEVELOPMENT OF INSTRUMENTATION AND CONTROL SYSTEM ARCHITECTURE FOR KNGR

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The reason for the transition to digital I&C systems lies in their many advantages over existing analog systems. Digital electronics are essentially free of drift that afflicts analog electronics, so they maintain their calibration better. System performance is improved in terms of accuracy and computational capabilities. Data handling and storage capacities are higher so operating conditions can be more fully measured and displayed. Properly designed, they can be easier to use and more flexible in application. Indeed, digital systems have the potential to improve capabilities (e.g., fault tolerance, self-testing, signal validation, and process system diagnostics) that could form the basis for entirely new approaches to achieve reliability [1].

With these potential advantages, a general shift to digital systems and waning vendor support for analog systems, the standard design for the Korea Next Generation Reactor (KNGR) has been undertaken since 1995, based on the conceptual design accomplished during 1992-1994. The preliminary design documents have been issued and standard documents, including both system design and design reports, will be completed by 1999.

The KNGR Instrumentation and Control (I&C) system has some design features, such as proven digital computers and networks, more advanced than those of nuclear power plants in operation in Korea. Advanced data communication technology in I&C systems necessitates not only the integrity of the whole system but harmonious communication between interfaced intra-systems.

The network-based system design demands close interaction among design teams such as the MCR (Main Control Room) design team, fluid design team, etc. The I&C structures of the Korean Standard Nuclear Plant (KSNP), ABB-CE Nuplex 80+, Edf N4, and Westinghouse AP-600 were reviewed and referenced to design the generic KNGR I&C system architecture. Moreover, the EPRI URD (Utility Requirement Document) was used for design bases.

The KNGR I&C system is composed of four quadrants. The first is the information systems important to safety, such as QIAS (Qualified Indication & Alarm System) and PAMS (Post Accident Monitoring System), necessary for monitoring plant conditions and safety system performance, and deciding preplanned operator response to events. The second comprises a non-safety information system such as IPS (Information Processing System) with a highly reliable client-server structure equipped with graphic-based workstations and servers to improve plant operability. The third is the plant protection systems such as Core Protection Calculator (CPC), Reactor Protection, and Engineered Safety Features Actuation Systems (ESFAS). The fourth includes control systems that function manually and automatically to maintain process variables within prescribed normal operating limits.

This paper describes license requirements such as defense against potential common-mode failures, software verification and validation, hardware qualification, and I&C reliability [2].

In this respect, this paper exhibits experience from the I&C prototype, composed of ten-system prototypes. This paper also shows design improvement made over the Power Control System (PCS), Plant Protection System (PPS), ESFAS, and other systems.

CHARACTERISTICS OF KNGR I&C

The main control room (MCR) of the KNGR adopts compact workstations with a large panel in front providing key plant operating status displays with critical function and success path monitoring capabilities, a computerized operation procedure, and one common soft controller. This controller is co-operated with the IPS information and gateway database from safety and nonsafety control networks and controlled by an operator through the FPD (Flat Panel Display) in MCB [3].

In accord with new MCR design concepts, the I&C design has the following characteristics:

- Application of full digital technology to the protective and control and monitoring systems, enabling performance of in-service auto-tests, self-diagnosis, and many flexible applications.
- A network-based structure making possible the integrated common I&C architecture between NSSS (Nuclear Steam Supply System) and BOP (Balance Of Plant). The architecture is composed of four segregated quadrants; these are safety control and information networks and nonsafety control and information networks.
- Functional segmentation to avoid loss of monitoring or control of the intra-function segments provided by the mechanical portion of the system. A multi-loop controller takes charge in one functional segment, instead of several single loop controllers for as many control loops.
- Use of high fidelity multiplexed fiber optic/coaxial cable data transmission including signals from sensors, operator consoles, switches, etc., and signals to actuators, relays, displays, etc., minimizing cabling costs and enabling electric isolation.
- Standardized modular but flexible designs to improve cost-benefit, to accommodate design changes and enhance the ability to easily replace equipment.
- Some changes in implementing technology compared with existing plants in Korea such as: adoption of a combined single processor to perform bi-stable functions and local coincidence logic in the PPS (Plant Protection System), full two-out-of-four logic, instead of selective two-out-of-four logic in ESFAS (Engineered Safety Feature Actuation System), and more compact advanced design in PCS to resolve inner noise problems in the power cabinet using microcomputers and the addition of frequency load following capability.
- Resolution of obsolescence problems the in CPC System with the design change of 2 CEACs (Control Element Assembly Calculator) to 4 CEACs and ICCMS (Inadequate Core Cooling Monitoring System). To upgrade these systems, software verification and validation and hardware qualification are underway.
- I&C Diversity Accident Analyses (qualitative analyses of 28 design basis events using Best-Estimate Evaluation Methodology and additional quantitative analyses using computer codes) were conducted to show diverse mitigation capability in case of common mode failure of the protection system.

The nine quantitative analyses events considered up to now are increase in feedwater flow, steam line break, (non-LOCA), loss of flow, seized/sheared shaft, CEA ejection, letdown line break, steam generator tube rupture, feedwater line break, and small break LOCA.

CONCLUSIONS

With the potential advantage of digital electronics, the standard design for the Korea Next Generation Reactor (KNGR) has been underway since 1995, based on a conceptual design from 1992-1994. It uses proven digital computers and networks for protective and non-safety systems; multi-loop controllers functionally segmented for reliability and economics, instead of several single loop controllers for one functional segment; high fidelity multiplexed fiber optic/coaxial cable data transmission to reduce cabling cost and for electric isolation; and standardized modular flexible design for easy equipment replacement. It up-grades existing systems by: adopting a combined single processor for bi-stable functions and local coincidence logic in PPS; the full two-out-of-four logic instead of selective two-out-of-four logic in ESFAS; the addition of frequency load following capability and compact design to resolve inner noise problems in power cabinets in PCS; and a design change from 2 CEAC (Control Element Assembly Calculator) to 4 CEAC in CPC Systems.

To resolve licensing problems arising from the adoption of digital technology, I&C Diversity Accident Analyses (qualitative analyses of 28 design bases events using Best-Estimate Evaluation Methodology and additional quantitative analyses using computer codes) have been conducted to show diverse mitigation capability in case of a common mode failure of the protection system.

For the validation of the functions of new design features in PPS, ESFAS, CPCS, PCS, network architecture, IPS, QIAS, and ICCMS, prototyping has been performed since 1996 and will continue before issuing the detailed I&C system design. Through detailed system design, the I&C design specification will be determined and detailed I&C function/logic drawings, including interconnection diagrams issued. See Figure 1.

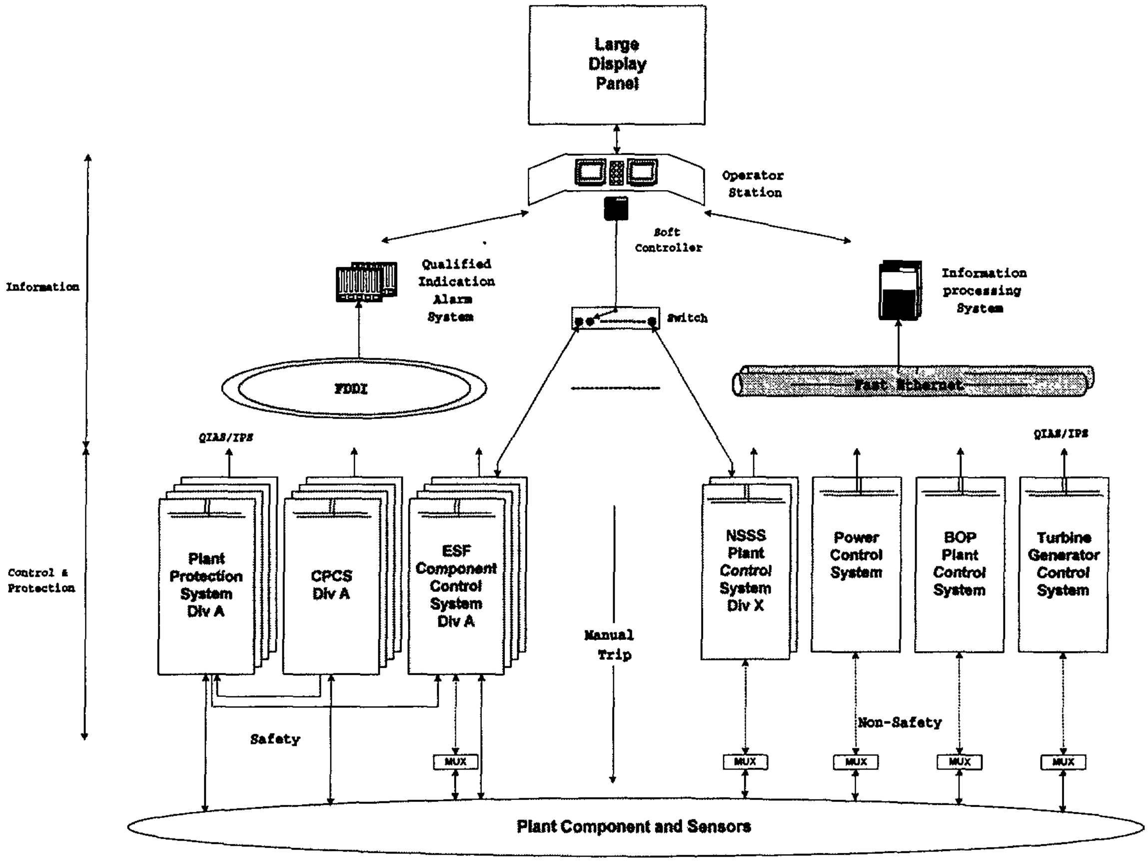

Figure 1. Overall I&C Architecture of KNGR.

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DEVELOPMENT OF HUMAN FACTORS EVALUATION TECHNIQUES FOR NUCLEAR POWER PLANTS

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This paper describes development of an operator task simulation analyzer and human factors evaluation techniques performed recently at Korea Atomic Energy Research Institute. The first is the SACOM (Simulation Analyzer with a Cognitive Operator Model) for the assessment of task performance by simulating control room operation. The latter has two objectives: to establish a human factors experiment facility, the Integrated Test Facility (ITF), and to establish techniques for human factors experiments

1. DEVELOPMENT OF ITF AND EXPERIMENTAL TECHNIQUES

Our research was performed in two themes. One, to establish an environment in which we can perform experiments for the human factors evaluation of human-machine interface (HMI) designs in nuclear power plants. Another, to perfect techniques for performing the experiments.

Advanced control room designs in other countries were reviewed for common features and the following were found: compact workstations using visual display units (VDUs), large overview displays, computerized alarm systems, a reduced number of operators, intelligent operating support systems, more automation than in conventional designs and information multiplicity. The requirements described in various documents for advanced man machine interface system (MMIS) designs were analyzed to derive items for experimental evaluation. Considering these items together with the common features, design factors were derived for the experimental evaluation of large overview displays and operator workstations. We surveyed the experimental evaluation in nuclear fields and general human factors experiments to classify the measurement methods. Then, experimental data collected by using the methods were categorized into system data, performance data, physiological data, and subjective data. The type of data and measurement devices effective for the evaluation of design factors were identified and related to each design factor. Also, three types of experimental facilities, a full-scope training simulator, a desk top system, and an integrated human factors test facility, were evaluated for advantages, disadvantages, and suitability for experiments on the items selected for evaluation. Based on these analyses, we defined basic components of ITF and their requirements to include a full-scope simulator with experimental control and data acquisition functions, VDU-based operator workstations with a large display, and operator performance measuring systems. The details of ITF were designed since 1995, and the test of ITF was completed in March of 1997.

ITF consists of KAERI-HMS (Human Machine Simulator) and experimental measurement systems. The KAERI-HMS includes a full-scope PWR-type nuclear power plant simulator with many VDU-based workstations. ITF has three rooms: the main test room (MTR), supporting test room (STR), and experimenter control room (ECR). Figure 1 shows the configuration of ITF. In the MTR, there is a VDT-based operator station, shift supervisor station, large-scale display panel (LSDP) and experimental measurement equipment. The reactor and turbine operator at the operator station obtain information from hierarchical plant schematics, alarm screens and trend graphs from the HMI of the terminals.

Operators can use touch screens, keyboards, mouse and trackballs to navigate among displays and to control components. The LSDP is in front of the operator station for information on plant status for the operators. The shift supervisor station has the same features as the operator station without component control functions. The STR can be used for subject training and pilot experiments can be

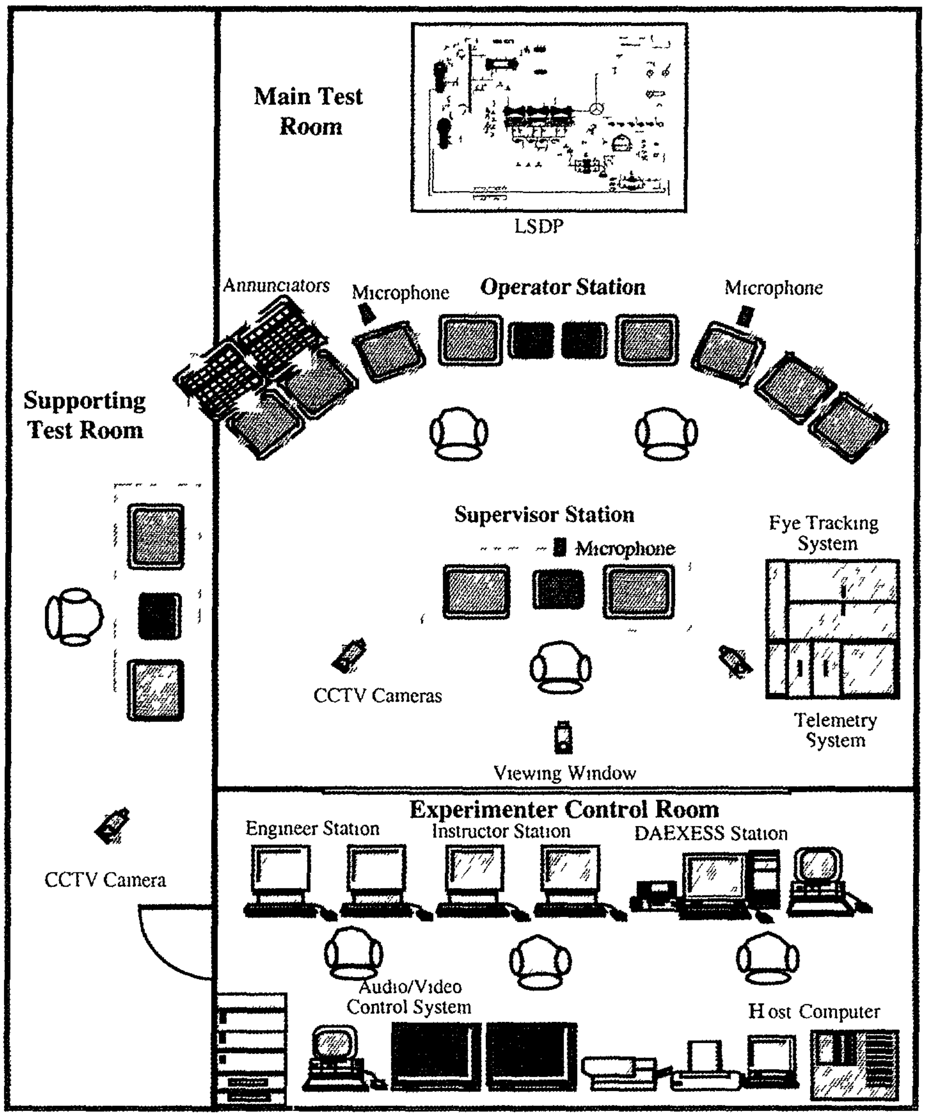

FIG. 1. ITF configuration.

performed independently and simultaneously with the experiments conducted in MTR. Even more, operations are possible with a new plant model or a new HMI design developed at the engineer station in the ECR.

In the ECR, there is a host computer with plant simulation and other necessary software and peripherals, an instructor station (IS), an engineer station (ES), a DAEXESS (Data Analysis and Experiment Evaluation Support System) workstation, and audio/video control system. Most experimenter functions are integrated in IS so he can activate desired functions and monitor experimental status through graphic user interfaces in IS. Menu-driven popup windows make it easy for the experimenter to select initial conditions, malfunctions and plant parameters to be monitored, set simulation speed, change plant parameters directly, catch snapshots of current status, backtrack, freeze, restart, replay or reset simulation, keep track of simulation status, report on-line logs, etc. IS can access all displays of the operator station of MTR during experiments. The ES has functions to maintain KAERI-HMS or design new human machine interfaces to be implemented later in MTR. Only ES can access all the modules of the simulation model, database, and graphic user interfaces and manage all simulation source code changes. In addition, to keep KAERI-HMS safe and reliable, the ES duplicates all the functions of the IS. The DAEXESS workstation is a system for experimental data analysis. The

HMI software, simulation model, and instructor software of the host computer produce the operator action log, alarm event log and plant parameter history, as shown in Figure 2. Then, SCADA (Supervisory Control and Data Acquisition) collects and stores the data and synchronizes physiological measurement data and eye movement records with them. The DAEXESS station receives the data from the SCADA and integrates them with audio /video record data. With the integrated data, experimenters can perform analyses, such as debriefing, statistical, operator performance, etc.

2. DEVELOPMENT OF SACOM

It was intended to develop a simulation tool of the operators' task for the dynamic human performance assessment of the operations in the main control room of a nuclear power plant. The operators' performance in the main control room operation is important to maintain the safety of nuclear power plants, especially during abnormal and emergency situations. To investigate the characteristics and

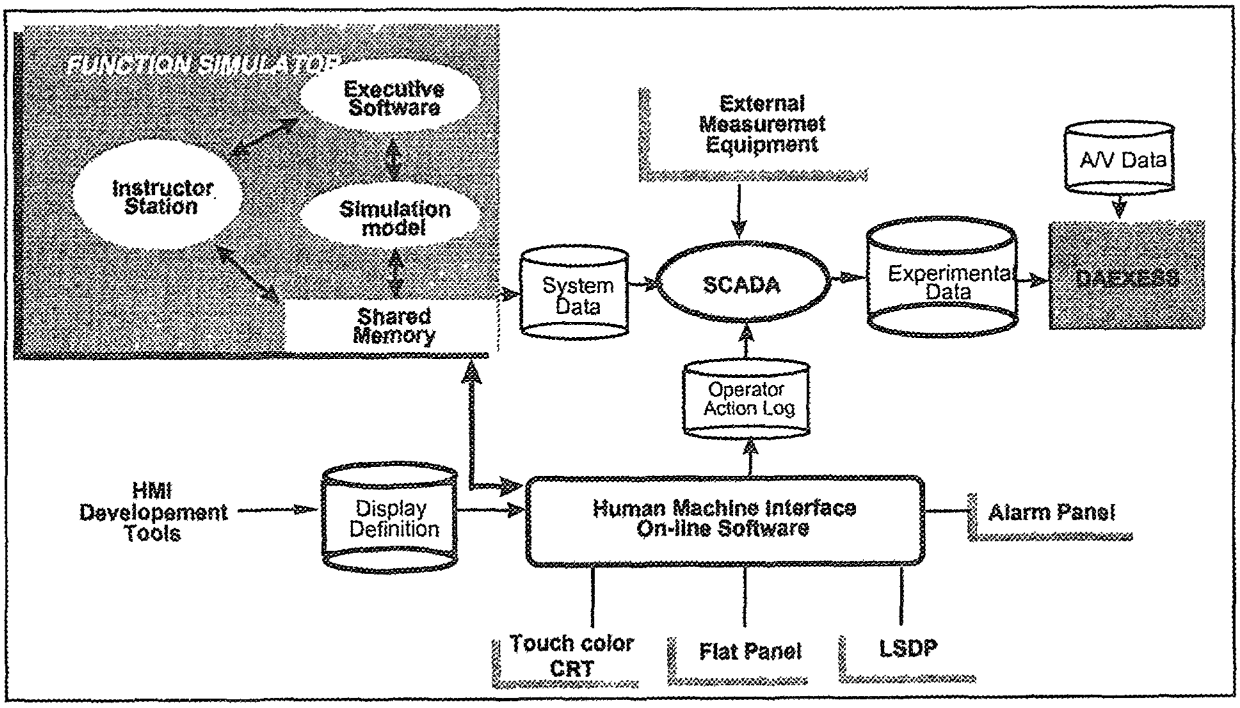

FIG. 2. Experimental data flow.

frames of operator tasks, we first analyzed operators' performance at a full-scope training simulator running several selected emergency operation scenarios. Before the simulator experiments, we investigated the requirements of tasks that operators should perform in accordance with the scenarios based on mainly procedures. Then, the operators' performance in the simulator experiments was captured through audio- and video- protocol, and reviewed in comparison with the requirements and frames of the tasks. This review was performed together with the operators who participated in the simulator operations and other operation experts (we call it "debriefing "). Based on the general cognitive mechanisms from literatures and the results from the simulator experiments, the base cognitive model of SACOM was established initially, and modified gradually with some plausible evidences obtained from the succeeding performance investigations.

Figure 3 shows the functional structure of SACOM. SACOM has three major modules: an operator model, an interaction analyzer, and a situation generator. The operator model simulates operator responses to given situations. According to the perceived situation, it retrieves knowledge from

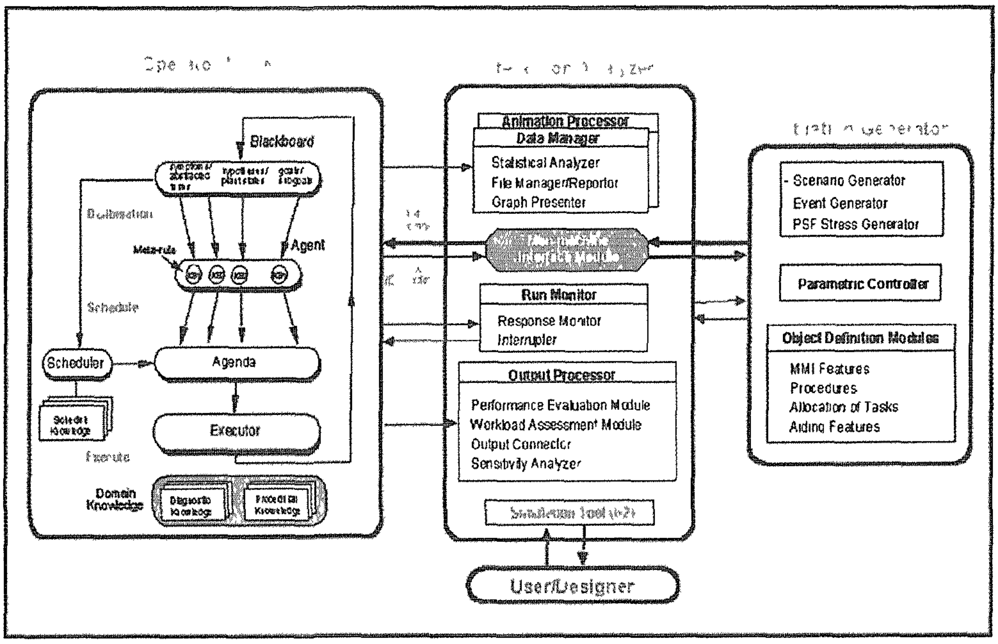

FIG. 3. Functional structure of SACOM.

the domain knowledge base, evaluates that knowledge, then issues responses to the interaction analyzer. The interaction analyzer transfers information between the operator model and the situation generator. This module displays the operating status of SACOM, performs the intended assessments, and displays the results. The situation generator changes the plant situation and handles events in the scenarios. Situation tables are used for this purpose at present. There are three kinds of user interfaces on SACOM for setup and data input, run control, and result processing, respectively. A prototype of SACOM for emergency operation scenarios was implemented on a black-board architecture and G2, which is

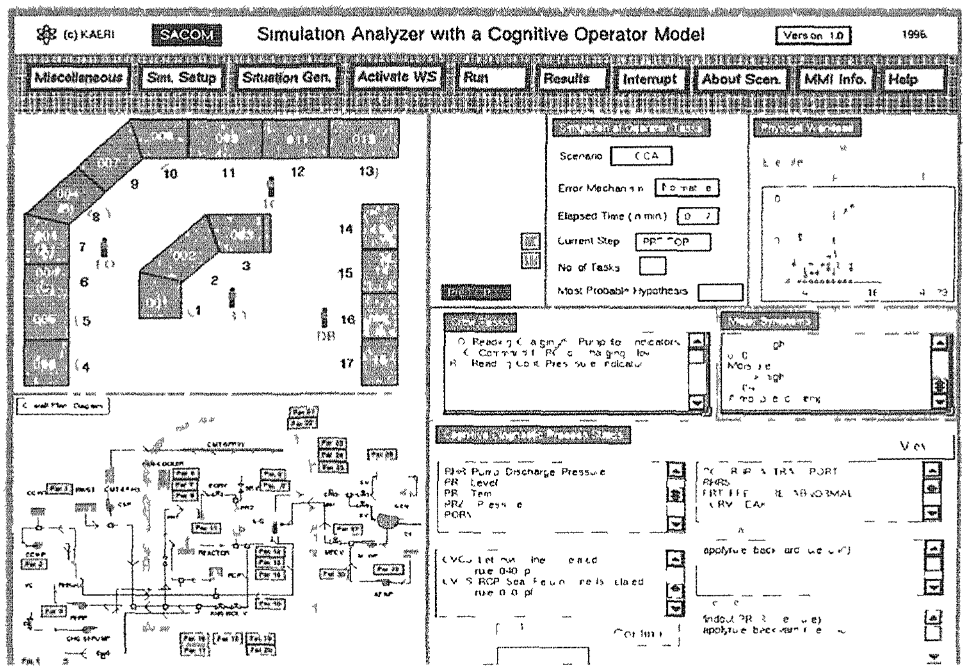

FIG. 4. An example screen of SACOM user interfaces.

amenable to update the knowledge base for other operational situations and the operator model through the more sophisticated cognitive theory.

For development of the framework of SACOM, we focused on human factors evaluation of HMI designs in both conventional and VDU-based types. With the SACOM, we can estimate the information requirements of tasks as primary measures and the cognitive workload and physical workload. We developed a new cognitive task analysis method for the derivation of task requirements and the assessment of cognitive workloads based on the concepts of cognitive span, working memory relief point, and working memory load. Therefore, SACOM includes the assessment functions to evaluate the arrangement and layout of control panels and devices, the quality of the information structure compared to task requirements, the suitability of function allocation and procedures, and error potential. Figure 4 is an example user screen when comparing the design alternatives for panel layout, or finding human factors problems in a conventional main control room design.

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NPP CERNAVODA UNIT 2 ECONOMIC VIABILITY — THE CHALLENGE FOR AN ADVANCED FINANCING SCHEME

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1. THE HISTORY

In the 1970's, Romania started an ambitious nuclear power program with the intention of building five nuclear power units at Cernavoda. To fulfill this program, the CANDU technology developed by Atomic Energy of Canada Limited (AECL), based on natural uranium as a fuel and heavy water as a cooling agent and moderator, was adopted. In December 1978, the CANDU technology procurement contract and other contracts were signed with AECL, through which the Canadians provided engineering and technical assistance, equipment and materials procurement from import, for the nuclear part of Unit 1. The services, engineering and procurement contracts were extended in 1981 for Unit 2. The site works started in 1980.In February 1981, a contract for the conventional part of Units 1 and 2 (the turbo-generator and its auxiliaries) was signed with General Electric (USA) and Ansaldo (Italy) companies.

The nuclear power program initially consisted of four power units of 700 MWe each, later, in 1982, it was extended to five units. Romanians managed the work on site, technically assisted by the Canadians and Italians while the State Commission surveyed the quality assurance and nuclear safety requirements for Nuclear Activity Control (CNCAN).

After 1989, the concept of the project development changed. Studies to establish the energy development and reorganization program concluded that completion of Cernavoda Power Plant was a necessity. Work was restarted under the new conditions, by implementing a thorough program of control, checks, and reparations for all previous work.

In July 1990, upon approval of the technical-economic study, the Romanian Government issued Decision No.750 for the project of "Cernavoda NPP 5 x 700 MWe". Also in 1990, a PRE-OSART mission invited by the Romanian Government, representing the International Atomic Energy Agency (IAEA), inspected the Cernavoda site. The mission made recommendations and suggestions, among others, to transfer the project management to a specialized company, highly experienced in nuclear plant construction.

As a result, in August 1991, a new contract was signed between the Romanian Electricity Authority (RENEL) and AECL – ANSALDO Consortium (AAC) to speed up and complete works on Unit 1. The object of the contract was to permit a take over of the management of work on Unit 1 from RENEL by the Consortium (AAC), to complete commissioning and to operate the Unit under foreign authority for an 18 month period. Also in 1991, to conserve financial resources and focus effort, it was decided to continue work only on Unit 1, temporarily discontinuing work on the other units. Until 1998, only preservation and some remedial and progress work were performed at Unit 2. Unit 1 was successfully put into commercial operation on December 2nd, 1996, and on September 30th, after one year and 10 month of operation, has an availability of about 85%.

2. UNIT 2 STATUS IN 1991

Unit 2 status in *1991* was as follows:

a) The design was complete and approvals required for 1991 construction were obtained from the regulatory authority.

b) About 70% of the equipment and materials, worth about 545 millions US $ were procured as follows:
   - 206 mil. US dollars import (mainly from Canada and Italy for NSP and Turbine Building);
   - 49 mil. US dollars import from General Electric (USA, for turbine generators);
   - 290 mill. US dollar local supply

c) Construction erection progress: 25%
   - 65% civil: main buildings and structures erected; only some internal structures, repairs and finishing to be completed;
   - 15% mechanical: few equipment (calandria, steam generators, pressurizer) and piping installed, mainly in turbine building;
   - 3% electrical and I&C

3. ECONOMIC JUSTIFICATION OF THE DECISION TO COMPLETE UNIT 2

To support the decision to complete Unit 2, Cernavoda a re-evaluation of the project was initiated:

- a detailed assessment of the Unit 2 safety design guides, design requirements and engineering design solutions in the light of changes in codes/guides/standards/actual regulatory requirements;
- identification of changes possible to implement, to meet the actual technological and regulatory requirements;
- Unit 2 cost/benefit analysis to demonstrate the economic efficiency of the project; and
- inputting costs for all electricity generation alternatives (nuclear, coal, hydro-carbon, hydro), versus forecast electricity demand into the "Least cost power and heat generation capacity development study" to select the best development plan for the Romanian utility.

3.1. Assessment of the Unit 2 safety design guides, design requirements and engineering design solutions in the light of changes in codes/guides/ standards/ actual regulatory requirements

A joint team RENEL-AECL-ANSALDO performed this evaluation, considering:

- Cernavoda Unit 1 as the reference for licensing, design, construction, organization of the project;
- Experience during construction and feedback from commissioning and operation of Unit 1;
- Experience from other Candu 6 NPPs in construction and operation;
- Actual codes and standards; existence on the market of the original equipment suppliers; their availability to provide support (repair, spare parts); equipment obsolescence (especially on electrical and I&C; e.g., station computer, power centers, shut down systems panels, control equipment etc);
- Current Romanian regulatory requirements.

To define the design applicable to Cernavoda Unit 2, we have used Unit 1"as built" documentation frozen at 30.06.1997, amended with a number of necessary changes identified or to be identified to be implemented and assessments to be done.

Significant specific issues of this activity are the following:

a) Assessment of the effect of aging on the existing equipment and structures.

b) Compliance with new AECB Canadian regulatory guides for containment, shut down, emergency core cooling systems and with reporting requirements for CANDU NPP (Wolsung 2-4 experience was used). All design changes implemented by AECL to up-grade Candu 6 design, taking W-2 as reference, were available. This list was revised and design changes for licensing and safety requirements were assessed for Cernavoda 2. In this assessment, design changes to ensure full compliance with AECB guides were considered and implementation proposed where feasible.

c) Applicable editions of the safety guides/standards/codes: it is required to use the latest editions except for ASME 1971, 1974 and 1980, and for ANSI 1971, 1979 and 1980 editions, as applicable.

The safety guides will be revised based on AECL experience with W2-4 to reflect AECB requirements with restrictions imposed by the project status;

To ensure compliance with 1989 Canadian standards, NSP systems have been reviewed in detail. This review did not lead to important design changes; in most parts, the latest revisions of CSA standards establish actual industry practices, or there was enough conservatism in the design to comply with more restrictive requirements. Because it is impossible to review the design for a revision date later than 1989, we would consider the following approach:

- the design will comply with the 1989 revision of Canadian norm and standards;
- the equipment already procured will meet the requirements of the standards in force at the time of procurement; and
- all new equipment will be procured in compliance with 1989 codes and standards, or more recent revisions.

d) Assessment before commissioning for common utilization of the water from the dousing bay and medium pressure stage of ECCS, common civil structures for Units 1 and 2 and compliance with single fault criteria defined for equipment:

Considering the tight completion schedule and cost restraints, we asked the regulatory authority for a deviation from the requirement to accept the completion of these assessments later, based on the safety philosophy used and accepted for Unit 1.

e) Evaluation of plant behavior during severe accidents (over design base accidents):

The expertise is unavailable. This requirement will be addressed as part of the long-term development plan, in cooperation with other Candu owners and in correlation with funds available.

f) Completion of level 1 PSA for all plant systems for nominal power, partial load and shut down status including external events before commissioning.

PSA level 1 for nominal power without external events will be produced and used like a good tool for design package verification. The rest will be addressed as part of the long-term development plan, in correlation with funds available.

g) A program accepted by the Regulatory Authority to prepare PSA level 2 and 3 in time for submission of the application for operation of Cernavoda Unit 2;

This requirement will be addressed as part of the long-term development plan in cooperation with other Candu owners, in correlation with funds available.

h) Assessment of the compliance with ISO 9000 for some manufactured equipment.

i) Total replacement of BOP analog/digital control system with a distributed control system.

j) Replacement of the station computers.

k) Improvements to the display/annunciation system in the plant control room, as well as greater attention to the ergonomics and human factors.
Improved construction sequencing to shorten the construction schedule.

3.2. Unit 2 cost/benefit analysis to demonstrate the economic viability of the project

To evaluate the capital cost required to complete Unit 2, all activities were identified to put Unit 2 in commercial operation: engineering, procurement, construction, project management, training, insurance, administrative and social requirements. Physical quantities applicable to the reference project and cost were assigned to each.

a) Engineering activities:
- Incorporating Unit 1 implemented modifications if applicable, in the Unit 2 design;
- Engineering work due to equipment obsolescence; and
- Producing the construction and safety design packages.

b) Procurement evaluation: the assessment balances material and equipment deemed necessary and with that already available from the original procurement for Cernavoda Unit 2, with the addition of equipment identified by the preservation group as unsuitable for use.

c) Construction assessment: man-hours required to erect the necessary physical quantities, based on approved Romanian Labor Standards, adjusted for local labor conditions and the experience gained during construction of the Unit 1. Ten per cent of the estimated construction time was added to deal with the envisaged non-conformities.

d) Training: Cernavoda NPP has a training center equipped with a full-scope simulator which, together with Unit 1 under operation, sets the necessary premises for the staff dedicated to Unit 2, providing adequate training according to international standards. The scope of work will represent the foreign instructor man-hours.

e) Insurance: includes the Construction All Risk (CAR) and the Nuclear All Risk (NAR) insurance.

f) Administrative and social costs to cover:
- an office building for project operations;
- refurbishing existing offices;
- refurbishing of the existing town site; and
- transportation facilities

g) Project management: mainly the manpower required on site to manage the project (site engineering, construction, procurement, material management, planning and scheduling, finance and administration), both Romanian and foreign specialists.

h) For each change required for implementation to meet the actual technological and regulatory requirements, an evaluation is performed to identify cost and schedule impact.

Capital cost to complete and put in operation Unit 2 is the sum of the above costs

3.3. Unit 2 cash flow analysis

A cash flow model has been developed. The financing scenario provides that all resources will be from loans, NUCLEARELECTRICA providing only interest during construction.

Results of the analysis show that Cernavoda Unit 2 is economically feasible. However, capital cost is a constraint that, if exceeded, may significantly decrease economic viability. A key factor is the ability of the owner to introduce competition into the procurement, management and construction of the project. Recent developments in the capital market put additional constraints on the economic efficiency.

3.4. Least cost power and heat generation capacity development study

In parallel, selection of the best development plan for the Romanian utility has been contracted to SEP - EDF - Tractebel consortium for the "Least cost power and heat generation capacity development study" based on construction and generation costs for all electricity generation alternatives (nuclear, coal, hydro-carbon, hydro), versus forecast electricity demand.

For 750 MUSD, capital cost to complete Unit 2, the least cost study shows that Cernavoda Unit 2 is the option ranked first. Sensitivity analysis shows that even for a higher than expected forced outage rate (15 versus the expected 8%) or for a higher (about 15%) than expected capital cost, Unit 2 remains attractive.

4. THE CHALLENGE FOR AN ADVANCED FINANCING SCHEME

In light of the extremely difficult conditions prevailing in the financing markets during the past few months, developing viable financing alternatives for the Project is a real challenge.

4.1. Risk analysis

A key issue for a successful financing plan is risk allocation. The risks in technology, construction, operation, fuel and heavy water supply, safety and licensing regulation and power sales have been considered.

Candu is a well-tested and mature technology. There are a lot of similar power plants in operation with satisfactory service records. Unit 1 in operation, with good performance records, represents a strong base of knowledge for the construction of Unit 2. We presume that the constructor will assume all risks associated with construction

As Unit 2 operator, Nuclearelectrica is responsible for the achievement of a minimum annual production to generate necessary revenues to reimburse the investors.

Based on Unit 1 experience, the Romanian manufacturers for nuclear fuel and heavy water are expected to supply for Cernavoda 2.

To overcome potential problems, we intend to clearly follow all the requirements of the Regulatory Authority before re-start of the works.

Nuclearelectrica will sign a power sales contract with CONEL to carry out transmission and distribution of electric energy. The possibility that CONEL will collect the bills and limit the enterprise arrears is a problem still to be addressed.

4.2. Potential financing schemes

To develop a viable financing structure, different alternatives must be considered, including: public markets, banks (syndicated loan), export credit agency (ECA), private placement and strategic partners.

Assessment of market conditions for these alternatives shows:

- the capital markets are extremely difficult without a sovereign guaranty;
- country limits are a constraint;
- financing with a major ECA depends on an upturn in Romanian and Eastern European markets generally;
- investor appetite for nuclear risk is limited;
- many key institutions are not interested or are restricted from investing in nuclear projects;
- investors require a premium return relative to either public markets or strategic partners; and
- a limited number of potential partners.

Most likely the final financing scheme will combine these alternatives to minimize, as much as possible in the actual framework, the State sovereign guaranty and the financing costs.

5. CONCLUSIONS

The completion of CERNAVODA Unit 2 is no doubt, an ambitious and challenging undertaking. The viability of the Project, under certain circumstances, has been analyzed and is strengthened by the following:

- The Project is grounded on a very positive background and environment, considering efforts of the Romanian Government to develop a comprehensive nuclear power program, the work already carried out and the infrastructures effectively in place. The best utilization of these infrastructures, with Unit 2 completion, will have a substantial positive economic impact on the operating costs of CERNAVODA NPP.
- The Project, viable in the light of Romania's Least-Cost criteria, might assume greater importance for the larger geographical area, considering the well-recognized urgency to shut down, as soon as possible, some older and un-safe nuclear power plants operating in neighboring countries.
- The main potential participants have successfully cooperated in the realization of Unit 1, and therefore, there are good premises for an effective organization to complete Unit 2 in the most favorable conditions concerning costs, technological performance and timing.

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XA0053600

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RUSSIAN LARGE POWER PWR: PRESENT STATUS AND PROSPECTS FOR THE FUTURE

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The Minatom enterprises conduct work to ensure the creation of the Russian large power PWR. It is performed according to the "Strategy of Russia nuclear power development" and oriented toward a wide application of the large power unit after 2010. The power use prognosis shows that even based on a conservative scenario, power use in Russia will increase 1.29 times by the year 2010 and 1.93 times by 2030 compared to 1995 levels.

The tendency during the last few years in Russia is to double-triple lead the rate of increase in fossil fuel cost compared to the rate of increase in nuclear fuel cost. NPPs with large power units are more effective than proposed fossil-fueled power plants in any European region of Russia. Factors reducing the prime cost of electric power and increasing the competitiveness of large power units are reductions in capital costs on the unit of the power generated during the plant lifetime, reductions in operating costs and reductions in staff.

A major portion of the demand for additional electric power is concentrated in the economic systems of the Center and the Urals (75-80%) during 2011-2030. At the same time, it is most effective to use NPPs with large power PWR in these regions (up to 20 units by 2030). These systems can be oriented towards larger power units because of present capabilities and the possibility of future development.

The safety design approaches of Russia and Europe for large-power PWR are practically identical: both designs qualify as evolutionary based on safety-related solutions, systems and equipment. At the same, time additional measures increase the reliability of safety barriers, including the double concrete containment.

When creating the large power PWR, the following priority problems are solved:

- increases in the unit power from 1300 MW(e) to 1500 MW(e);
- exclusion of premises and conditions for typical operational failures that occur (SG issues, FA deformation, CPS seizure, main circulating pump failures, etc.);
- increases in equipment and service life;
- application of technical solutions in safety, ensuring licensability of the units up to 2010 and later, concentrating on increased self-protectiveness and resistance to common cause failures: (including external and internal impacts and personnel errors) and measures to prevent and limit the consequences of heavy accidents;
- simplification of design and operation; and
- reduction of construction and operation costs.

The RF Ministry of Atomic Energy is interested in developing international industrial cooperation in the nuclear field and supports the aspiration of the Russian enterprises and institutes to participate in international projects on the development of new safer reactors.

Stage	Work participants
1. Conceptual design of units with about 1300 Mwe power 1994	General Designers:
2. Technical requirements for large power PWR (EUR analog) 1995	GNIPKII AEP, NIAEP, St-Peterburg AEP.
3. Calculation and design developments and technical and economic assessments of units of 1370-1500 MW(e) 1995-1996	Chief designers:
4. Analysis of design solutions and requirements to ensure correspondence to the requirements of EUR and western regulations (1st stage) 1996	OKBM, OKB "Hydropress", Scientific Supervisor of: RRC KI
5. Development of Reactor Island and Reactor Plant main equipment since 1996	
6. Development of alternative FA for VVER and large power unit since 1996	
7. Development of typical service equipment for VVER units and large power unit since 1996	
8. RP main equipment technical design (1st stage) 1997	
9. Research work: Analysis of competitiveness of large power atomic units and possible scales of construction on the European territory of Russia after 2010 1997	

BASIC DATA FOR NPP WITH UVR-1500 UNITS

Unit electrical output, MW	
gross	1500
net	1425
Number of units	2
Unit technical service life, yr.	60
Number of installed power operation hours, estimated by availability factor	7500
Design number (for competitiveness analysis) of installed power operation, hours	6500
House load, % from unit power output	5
Staff number per GW(e)	350
Decommissioning cost, % from capital costs	<40
Specific estimation capital cost relatively to that for PGU-CES	2.6
Owner's expenditures + uncertainties, % of capital cost	30
Refueling strategy, yr.	1x5
Average fuel discharge burnup, MWd/t	55,800
First core enrichment (average), %	2.4
Average equilibrium fuel enrichment, %	4.7
Average annual loading of new fuel assemblies	54
Specific natural uranium consumption per year, g/MWd (once-through cycle, 0.1 %	
U-235 content in depleted uranium)	168
Capacity factor	0.85
Availability factor	0.97

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NUCLEAR ENERGY DEVELOPMENT IN THE RUSSIAN FEDERATION: MODERNIZATION OF OPERATING AND NEW NPP CONSTRUCTION

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Basic target parameters of nuclear power industry development in Russia are [I]:

NPPs total installed capacity, GWe

Year	1998	1999	2000	2001-2005
Capacity	21,2	23,2	24,2	26,9
NPPs being commissioned		Completion of construction Rostov NPP (Unit.No.1) Kalinin NPP (Unit.No.3) Kursk NPP (Unit.No.5)		Rostov NPP (Unit.No.2) Novovoroneg NPP-2 (Unit.No.6) Sosnoviy Bor (Unit.No.1)* * - new design projects

The share of nuclear power for the period 2000-2005 will be 14%, at present it is-13, 6%

Economic aspects [1]:

Parameters	NPPs	Gas-fired Power Plants	Coal-fired Power Plants (ecologically clean)
1. Specific capital investments ($./kW)	1290-1470	570-730	940-1140
2. Fuel price ($/t.c.f**)	55-83	66-130	38-71
3. Specific expenses for service life (cents/kWh)	5,3-6,3	4,4-6,6	5,9-7,1

**t.c.f - ton of conventional fuel.

New safety requirements being implemented by the Russian Regulatory Body increase the role of the utilities. At the same time these requirements are evolving, resulting in the generation of a series of normative documents to operate NPPs. These and other factors necessitate elaboration and implementation of safety improvement programs with due regard for international practice.

Some measures for modernization of Kalinin NPP (Unit N3) are given below:

- implementation of a new program of RPV surveillance-specimens and development of a monitoring system for built-up fluence to define the RPV residual life;
- reduction in scram rod insertion time;
- implementation of the new refueling strategy (in/out) with neutron-low outflow;
- development and implementation of a system to monitor steam-gas under the reactor top head;
- containment hydrogen detection and recombination system implementation;
- water from ECCS heating-up;

- making steam lines and feedwater pipelines "rigid" at the reactor compartment outlet;
- development and implementation of measures to compensate ECCS heat exchanger loss due to obstruction with thermal insulation during LOCA;
- development and implementation of diagnostic systems; and
- increase of discharge time of batteries to 1 hour.

As seen from the Table, construction of new design NPPs is planned for the near future: VVER-1000 (design V-392) - Novovoroneg NPP-2 and VVER-640 (design V-407) - Head unit NP-500, town of Sosnoviy Bor. The main principle of design is an approach including technical solutions confirmed by positive experiences in NPP operation, including modernization.

In addition to research on high-power reactors (~ 1500 MWe) of the VVER-type, research on fast reactors is under way. For this, about $100 million/year is allocated for the entire scope of R&D until 2000. For a nuclear steam supply system with a high-power VVER reactor, some major parameters are as follows:

Parameters	**VVER-1500**	**VVER-1000**
Reactor power, MWt	4250	3000
Reactor vessel service life, years	60	40
Fluence of fast-neutron (E>0.5 Mev) to reactor vessel for service life, neutron/cm^2	1,7*10^{19}	5,7*10^{19}
Pressure above core, Mpa	15,7	15,7
Coolant temperature, °C:		
Reactor inlet	294,6	289,7
Reactor outlet	323,8	320,0
Reactor coolant flow rate, m^3/h	126760	84800
Steam generator outlet pressure, MPa	7,06	6,27
Number of fuel assemblies	211	163
Total number of fuel rods	93684	47759
Fuel rod outer diameter, mm	6-7	9,1
Core lifetime, years	4-5	3
Maximum fuel enrichment, %	4,4	4,4

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ECOLOGICAL AND RISK-RELATED PERFORMANCE OF NUCLEAR AND OTHER ELECTRICITY GENERATING SYSTEMS

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This paper presents some highlights from the ongoing Swiss project on comprehensive assessment of energy systems, carried out by the Paul Scherrer Institute (PSI) in cooperation with the Swiss Federal Institute of Technology, Zürich (ETHZ). Its objective is to develop, implement and use a comprehensive methodology for the consistent and detailed assessment of energy sources and thus provide scientific support to the decision-making process concerning the future configuration of the Swiss energy system. The types of energy sources covered include fossil fuels, nuclear and renewables.

The approaches used in this analysis include [1]:

- Life Cycle Assessment (LCA), primarily for the detailed analysis of environmental inventories; the results cover a large spectrum of gaseous and liquid emissions, wastes, as well as energetic and non-energetic resources (e.g. raw materials).
- Environmental and health impact assessment based on simulation of the transport, chemical conversion and deposition of major pollutants.
- Direct use of historical accident experience and applications of Probabilistic Safety Assessment (PSA) for the analysis of severe accident potential of the systems of interest. In this context a database having a superior coverage of energy-related accidents has been developed.
- Modeling of the economic impacts of environmental policies through application of a large scale "bottom-up" energy-economy model.
- Multi-criteria framework allowing to simultaneously address the often conflicting socioeconomic and ecological criteria.

In this paper the scope is limited to electricity generating systems; results examples are provided comparing LCA-based environmental inventories and risks associated with severe accidents. In both cases the assessment is based on the consideration of full energy chains. The LCA results (primarily based on [2]) reflect the impact of anticipated technical advancements in a 20 to 30 years perspective.

An example, illustrating the application of LCA to the advanced nuclear systems (ABWR and AP600 were used as the reference power plant technologies), is shown in Figure 1.

Figure 2 shows a comparison of normalised Greenhouse Gas (GHG) emissions for current systems operating in Western Europe or in Switzerland, and for future systems operated under Swiss conditions. Minimum and maximum values are indicated whenever applicable; the percentages represent range of reductions due to expected technology advancements, taking the present situation as the reference.

Advanced systems feature considerably lower specific emission levels than today's systems. In the case of Greenhouse Gas emissions reductions by up to a factor of two will be feasible for natural gas systems, and by a factor of between two and five for photovoltaic (PV) integrated slanted roof panels. For non-fossil energy chains, direct emissions from power plants make a negligible contribution to the total quantity. For atmospheric pollutants such as nitrogen and sulphur oxides, the

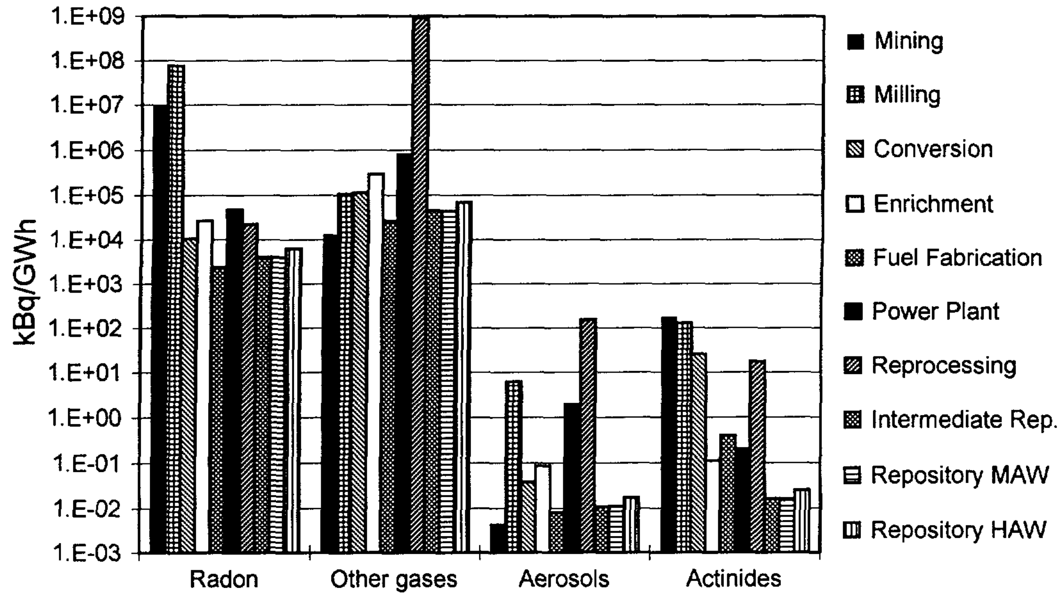

FIG. 1. LCA-based radioactive emissions to air from normal operation of future nuclear system.

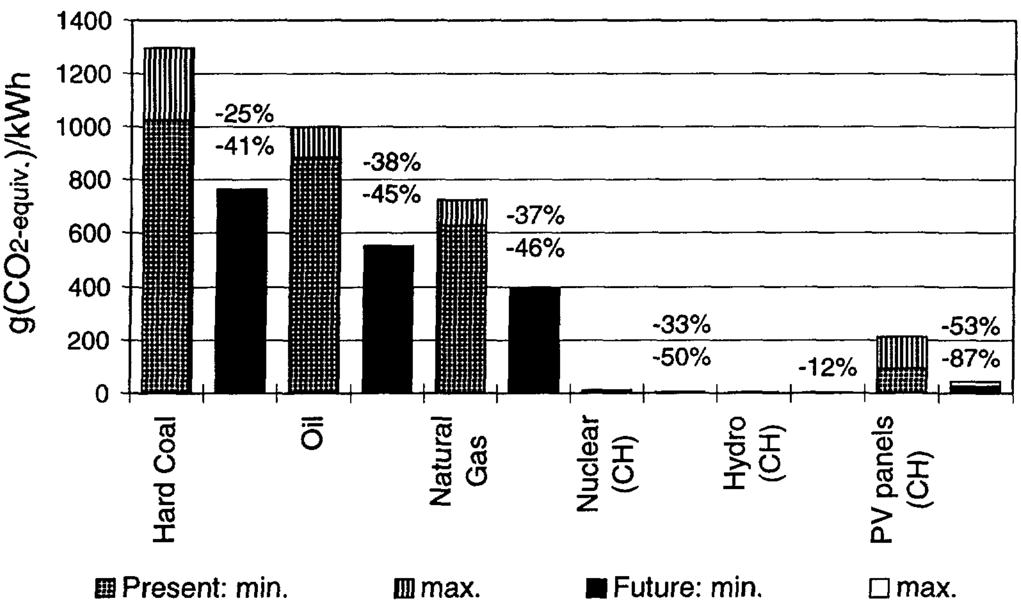

FIG. 2. Comparison of current and future LCA-based Greenhouse Gas emissions - impact of technology improvements.

LCA approach emphasises the increased relative significance of other stages in the energy chain (e.g., transport), and of indirect emissions (particularly in connection with the manufacture of materials).

For the purpose of comparison the risks of severe accidents associated with various energy chains can be expressed in terms of damage (e.g. number of fatalities) normalised by the electrical output (either direct or equivalent to the thermal energy). For extensive presentation and discussion of the comparative analysis, and the database behind it, we refer to [3]. Figure 3 shows the historically based estimated number of immediate fatalities per unit of energy for six energy chains. The considered evaluation period is in this case 1969-1996.

Distinction is made here between OECD- and non-OECD-countries. In the case of fossil sources the occurrence of severe accidents is concentrated to energy chain stages other than power plants (mainly extraction and transportation). Allocation of accidents was carried out utilising weighting based on import and export balances for specific energy carriers. It should be noted that delayed fatalities, particularly relevant for the Chernobyl accident, are treated separately in [3]. The same reference provides an extensive probabilistic treatment of hypothetical nuclear accidents, including the analysis of the corresponding external costs.

The differences between OECD- and non-OECD-countries are significant but not dramatic for the fossil systems. On the other hand, for hydro and nuclear systems, where the risks are concentrated to the power generating plants, the differences correspond to two orders of magnitude. In fact, the hydro estimate for the OECD-countries is comparable with the PSA-based estimate of $2.0*10^{-2}$ latent fatalities per GWe-year obtained for the Swiss nuclear power plant Mühleberg (there are no immediate fatalities for this plant).

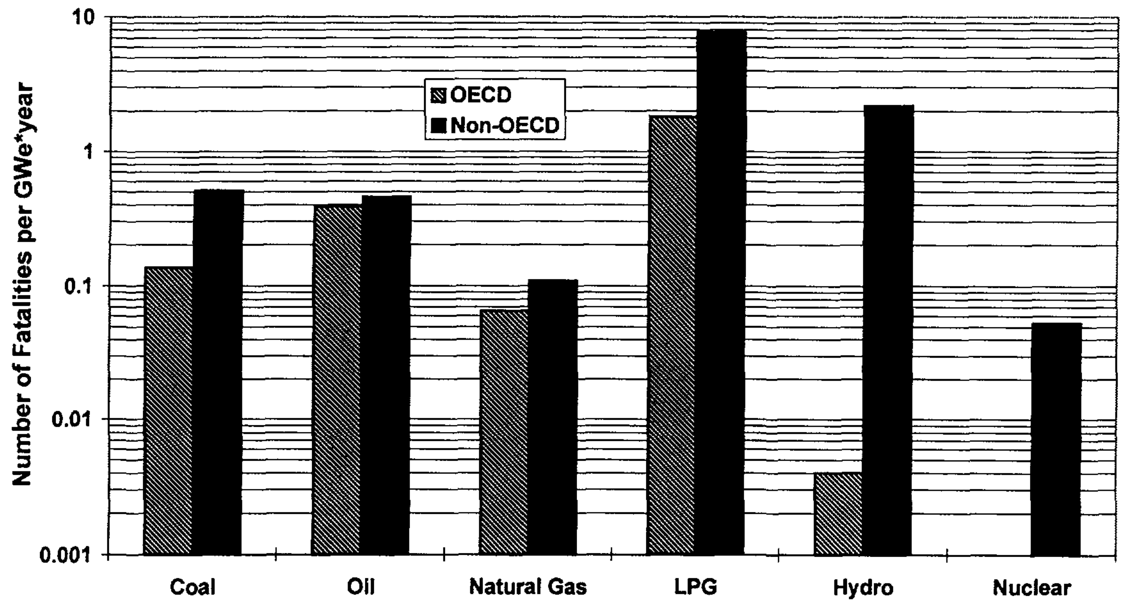

FIG. 3. Immediate fatality rates (number of fatalities per GWe-year) for different energy chains for the time period 1969-1996.

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WATER-COOLED REACTORS IN THE UKRAINIAN POWER INDUSTRY: OPERATIONAL PROBLEMS AND ECONOMIC IMPROVEMENT

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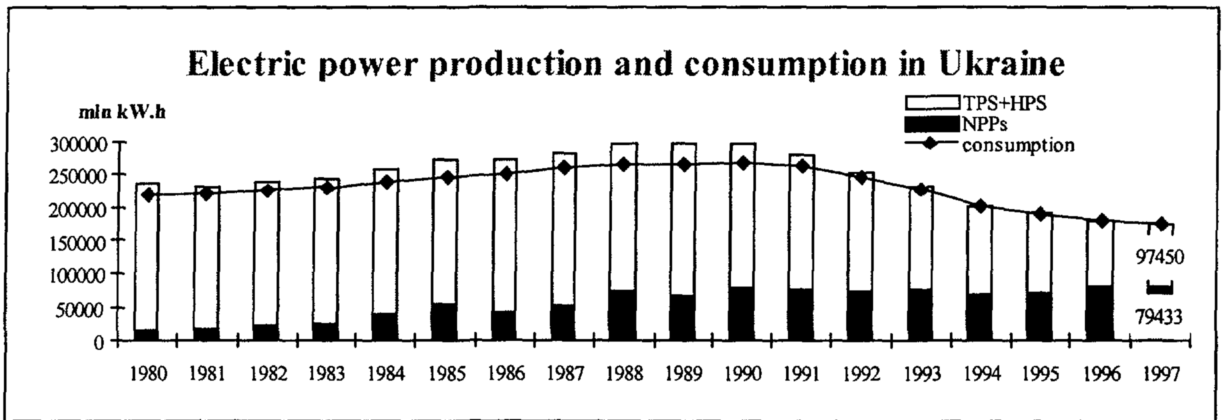
In Ukraine there are fourteen water-cooled power reactors in operation (11 VVER-1000, 2 VVER-440 and 1 RBMK-1000) with a total installed capacity of 12800 MWe. This is about 25% of the total installed capacity of the electric power stations. Four VVER-1000 units are under construction. In 1997 NPPs produced 44,9 per cent of the electricity in the country. For the last 15 years, there has been a preference for NPP units in power plant construction in Ukraine.

The share of electricity produced by NPPs constantly increases because of the economic crisis and cheaper electric power production by the NPPs (see Fig. 1.).



The VVER-1000 NPPs operation (120 reactor-years) has revealed the following problems:

- Insufficient reliability in the integrity of steam generator (SG) work. At 16 operating VVER-1000, steam generators cracks have developed in cold collectors during operation. In three cases, ligament integrity damages were indicated by increases in steam generator water activity. In all other cases, the damage was detected during annual scheduled in-service inspection. Fifteen SG were replaced; one was repaired at the plant, not replaced.
- Effective compensatory and interim measures (thermal heat treatment [at approximately 450^{0}C] of cold and hot collectors, feedwater, blowdown and steam separation modification) were implemented at all SG of operating plants and those under construction.
- Insufficient reliability of Instrumentation and Control Systems (I&C). A program to replace existing (I&C) systems with advanced systems is being developed and realized.
- In 1992-93, at almost all VVER-1000 of Ukraine, Russia and Bulgaria, incomplete RCCA insertion has been revealed (i.e., RCCA stuck in an intermediate position, RCCA drop time exceeds 4s - design time). A program of additional quarterly measurements of RCCA drop time was developed and implemented. When there is RCCA operational violation and no chance to stop it, the unit is transferred to the operational mode with three-loop coolant circulation and a preliminary power reduction to 67% of $N_{nominal}$.

The cause of incomplete RCCA insertion is fuel assembly (FA) bowing. To predict the water gap increase and avoid excess fuel rod linear power density, measurements were made of FA bow in the reactor core. Based on bow measurements and following calculations, operational restrictions (reduction of reactor power of 5 -10 %) were entered in 1996 -1997.

Some compensatory measures were applied to maintain the design RCCA drop time, i.e.:

- a modification upgrading the bundled safety tubes (BST), to correct BST position and to correct axial compression of FA;
- drilling of RCCA drivers bars to reduce the force of hydrodynamic friction during the RCCA input in the reactor core;
- using the RCCA and RCCA driver bar with an increased dead load;
- organizing the core loading pattern with advanced FA with zirconium (Zr -Nb and Zr - Nb-Sn -Fe alloys) guide thimbles and grids and with increased (compared with initial design) FA headspring gain.

The result of compensatory measures (excepting a few cases) is that an increase of RCCA drop time (above design time t=4 sec.) was eliminated.

In Figure 2 (an example of a Z NPP-2) the curves of RCCA drop time (sec.) change for the period 1993- I quarters 1998 are represented.

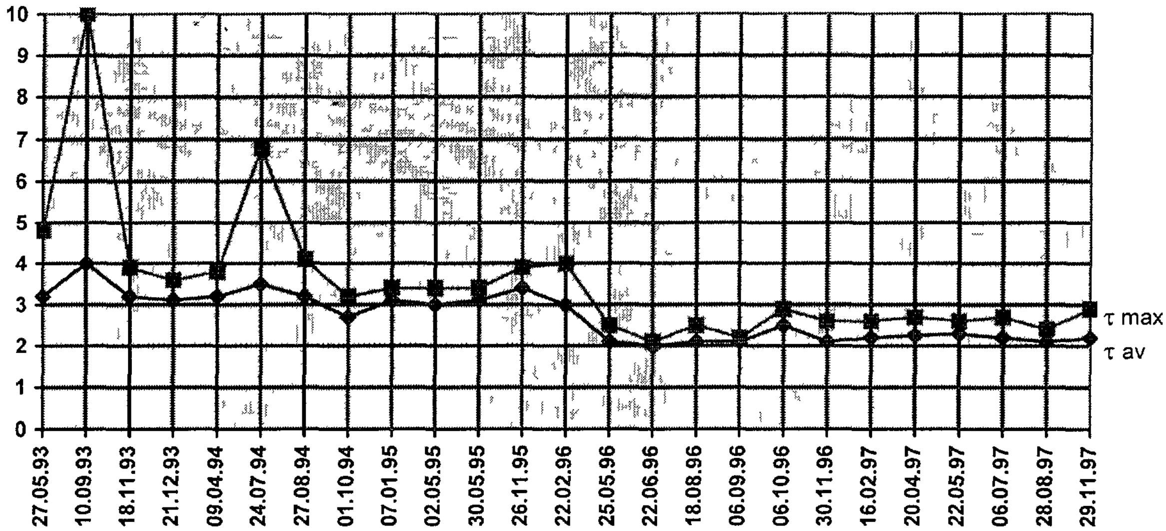


FIG. 2 Measurement of RCCA drop time on Z NPP2

The reliability of VVER-1000 fuel remains at the usual level (Figure 3). This supports the conservative approach during the operational restriction determination.

The increase of NPP with VVER profitability is provided by:

- an increase in fuel burn up using some FA of the 3 fuel cycles design in the 4th fuel loading cycle. Figure 4 shows burnup distribution in the unloaded assemblies, and
- introduction of loading patterns with reduced leakage of neutrons Average fuel burn up at Ukrainian NPPs with VVER-1000 is 4-10 % higher than that at NPPs with VVER-1000 in other countries. It makes possible a reduction in quantities of spent fuel and thus, fuel cycle back end costs

The substantial prolongation of the VVER reactor vessel lifetime (presumably, from 30 to 40 years) allows a reduction in the electricity investment and in the amount of spent fuel (because of lower fuel burn up in new unit initial core loading).

The inert radioactive gas release into the atmosphere (disposal stack) is usually not more than $1*10^{11}$ Bq /day per unit for the VVER-1000.

Part of the VVER spent fuel is shipped to Russian re-processing plants with the prospect of MOX fuel production and utilization. At the same time, a program of dry interim storage of spent fuel is being implemented in Ukraine.

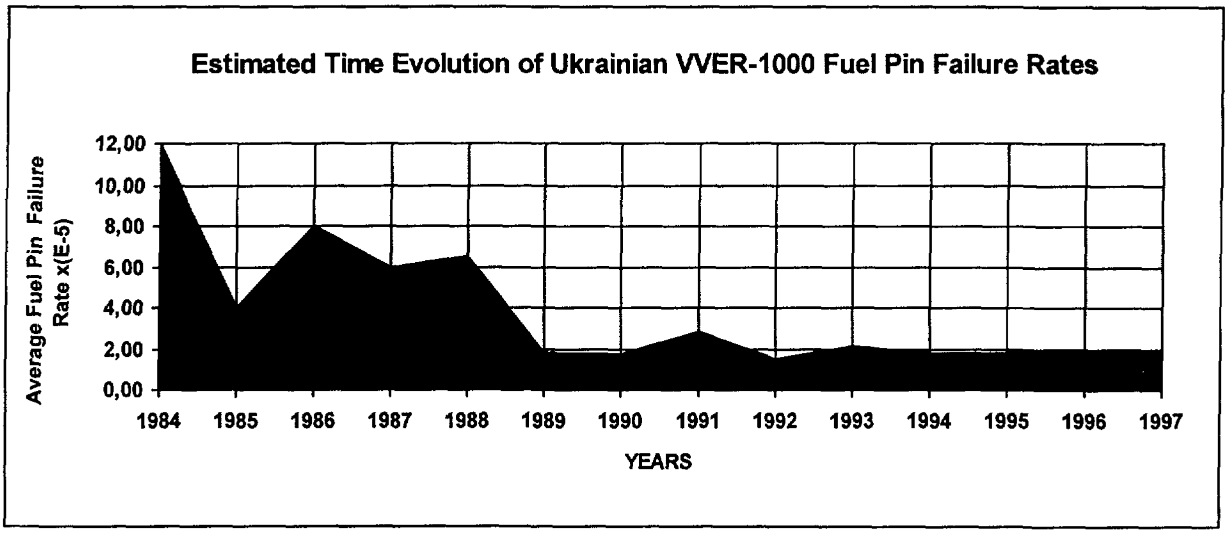


FIG. 3. Estimated time evolution of Ukrainian VVER-1000 fuel pin failure rates.

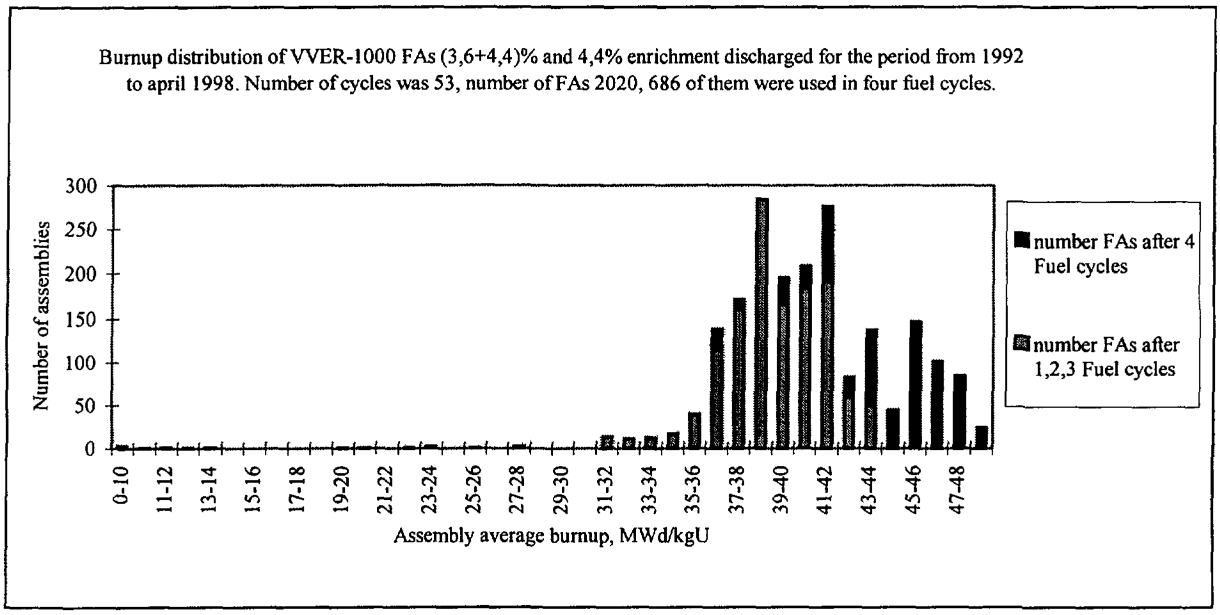


FIG. 4. burnup distribution of VVER-1000 FAs (3,6+4,4)% and 4,4% enrichment discharged for the period from 1992 to April 1998. Number of cycles was 53, number of Fas 2020, 686 of them were used in four fuel cycles.

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UTILIZATION OF RISK-INFORMED INSERVICE INSPECTION TO REDUCE O&M COSTS OF NUCLEAR POWER PLANTS

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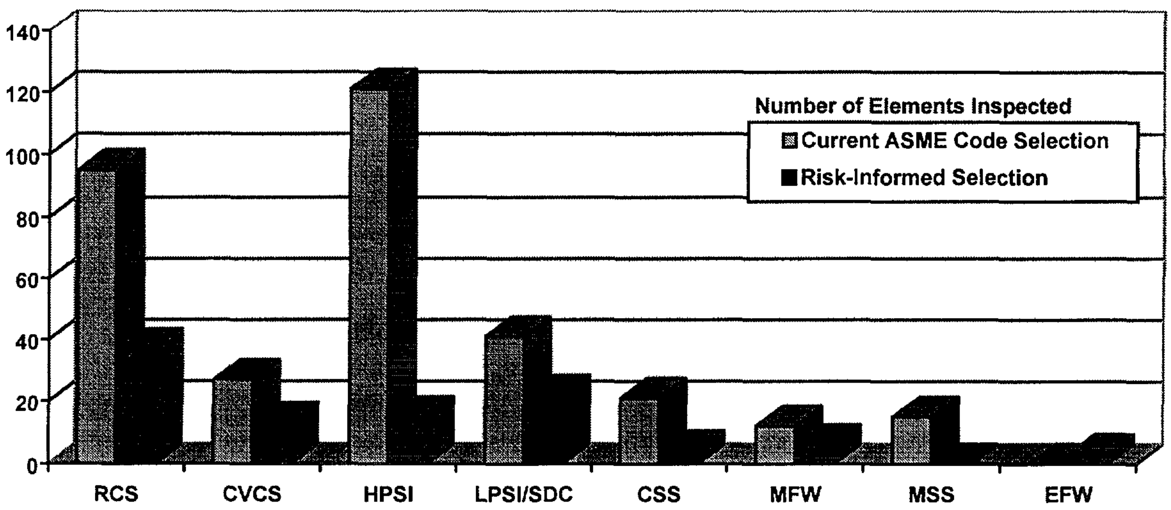
The Electric Power Research Institute's (EPRI) Risk-Informed Inservice Inspection Evaluation Procedure[1] has been used by ABB in a pilot application at Arkansas Nuclear One, Unit 2 (ANO-2). The risk-informed inservice inspection is used to identify risk-significant pipe segments and to define the locations that are to be inspected within these segments. A diverse set of systems comprising class 1, 2 and 3 piping, as well as selected non-code piping was included in this pilot application of the EPRI procedure. The systems evaluated included the Reactor Coolant System, Chemical and Volume Control System, High Pressure Safety Injection System, Low Pressure Safety Injection/Shutdown Cooling System, Containment Spray System, Main Feedwater System, Main Steam System and Emergency Feedwater System. A risk evaluation was performed by dividing each system into piping segments, each with a characteristic set of failure consequences and degradation mechanisms. Failure consequence and degradation mechanism evaluations were performed to assign each of the piping segments to one of seven risk categories, as defined in the EPRI procedure. This assignment provided a mechanism for evaluating the risk significance of each piping segment. Finally, the inspection locations were selected.

The consequence evaluation that was performed focused on the impact of each pipe segment failure on the capability of the affected system to perform its design functions and on the overall operation of the plant. Impact due to both direct and indirect effects were considered. A direct effect is generally a major perturbation in plant process parameters which necessitates immediate mitigating actions (automatic or manual). Indirect effects are generally those caused by flooding, spraying, and/or jet impingement resulting from the failure of pipe segments in neighboring equipment or interfacing systems. Determination of the consequences of a segment failure considers the potential of losing one or more trains of affected mitigating systems, and the consequential impact on safety functions. Several sources of information including the ANO-2 Individual Plant Examination (IPE) for internal and external events, Internal Flood Screening Study and various plant design drawings were used to perform the consequence evaluation. In addition to these sources, a plant walkdown for the various systems of concern was conducted. The walkdown captured subtle interactions which could not be readily identified using the other sources of information. Observations from the walkdown were factored into the consequence evaluation. Based on the consequence evaluation, each of the piping segments was categorized as having a consequence level of "HIGH", "MEDIUM", "LOW" or "NONE". The at-power plant configuration is considered to present the greatest risk for piping failures since the plant requires immediate response to satisfy reactivity control, heat removal, and inventory control. Although the consequence evaluation focused on the at-power configuration of the plant, shutdown operation and external events were also assessed to gain a level of confidence that the consequence ranking during these other configurations would not be more limiting.

The degradation mechanisms that can be present in the piping depend on design characteristics, fabrication practices, operating conditions and service experience. The degradation mechanism evaluation that was performed as a part of the pilot application addressed the potential for thermal fatigue, stress corrosion cracking, localized corrosion and flow sensitive attack in each of the piping segments. An exhaustive review of databases, plant specific and industry wide, was conducted to characterize ANO-2's operating experience with regard to piping pressure boundary degradation. This included a review of all historical water hammer events that have occurred at the plant, with consideration given to subsequent preventive actions (i.e., design changes, operational practice changes) taken, to assess the likelihood of potential future occurrences in conjunction with the presence of a degradation mechanism. In addition, several sources of information, including the Flow Accelerated Corrosion Prevention Program, the Primary Chemistry Monitoring Program and various operating procedures were used to identify potential degradation mechanisms for each of the systems considered. The level of severity for each degradation mechanism for each piping segment was categorized as "Large Leak", "Small Leak" or "None" depending upon industry historical experience with each mechanism.

The combination of the consequence and degradation mechanism categories produced piping runs, known as risk segments, each of which was categorized as having a risk level of "High", "Medium" or "Low". Each risk segment consists of a continuous run of piping, that, if failed, has the same consequences, and is exposed to the same degradation mechanisms. The selection of individual inspection locations within a risk segment depends upon the relative severity of the degradation mechanism present, the physical access constraints, and the radiation exposure. In the absence of any identified degradation mechanisms, the selections focused on terminal ends and other locations of high stress and/or high fatigue usage.

The pilot application of the EPRI Risk-Informed Inspection at ANO-2 [2], as shown below, resulted in a 70% reduction in the number of elements selected for inspection. All the systems considered, except for the Emergency Feedwater System, would experience a significant reduction in the number of elements inspected compared with the current inspection program under the ASME Code. This translates into considerable cost and personnel radiation exposure savings to the plant.



ANO-2 Before and After Element Selection Comparison

For a ten-year inservice inspection interval, plant O&M cost savings are conservatively estimated in excess of 1 million US dollars with a corresponding worker radiation exposure reduction of 75-100 man-rem. The greatest savings occurred in Class 1 piping systems due to more stringent Code requirements and adverse plant conditions. By focusing the inspections on the higher risk elements, we can reduce the number of inspections without compromising the effectiveness of the inspection program.

RCS	Reactor Coolant System
CVCS	Chemical and Volume Control System
HPSI	High Pressure Safety Injection System
LPSI/SDC	Low Pressure Safety Injection/Shutdown Cooling System
CSS	Containment Spray System
MFW	Main Feedwater System
MSS	Main Steam System
EFW	Emergency Feedwater System

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POSTER SESSION II

INTRODUCTION OF MEASURES TO ENHANCE SAFETY AT THE ARMENIAN NUCLEAR POWER PLANT

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The Armenian Nuclear Power Plant (NPP) is one of the plants built in the 1960s and 70s on the model of Units 3 and 4 of the Novovoronezh NPP, i.e., a first-generation plant with a WWER-440/V-230 reactor. The Armenian NPP site-specific feature is the possibility of a magnitude 8 earthquake (MSK 64 scale), which caused fundamental design improvements not only to construction, but to the whole reactor facility, and gained a new serial number, V-270. In the absence of State-wide regulations, a number of technical solutions differing from those applied at other WWER-440/V-230 reactors were adopted from the beginning:

- The reactor vessel was fastened to the foundations by a supporting structure;
- The steam generators, reactor coolant pumps and main stop valves were equipped with hydraulic shock absorbers;
- The pressurizer and other pumping and heat exchange equipment were fastened to the reinforced concrete structures in two horizontal planes;
- The GTsN-310 reactor coolant pump was replaced by the inertial GTsN-317 pump;
- The EhP-50 type emergency injection pumps were replaced by TsN 65-130 pumps;
- Two new cooling systems were installed: the ESP (emergency seismic pumps) and the ECP (emergency condenser pumps), located in the seismically-resistant building of the boron room;
- The spent fuel pool and boron solution tank were built with a double lining of stainless steel sheets;
- The reactor facility was upgraded by the seismic protection system;
- The refuelling machine and 250 t/s bridge crane (Reactor Compartment) were redesigned to be seismic-proof;
- 0.4 kV switchgears of the emergency power supply system were lowered from the +9.6 m elevation to the 0.0 m elevation;
- All 6 and 0.4 kV sections of the ASG (auxiliary switchgear), protection and control panels of all types were fastened at the top to reinforced concrete structures;
- Power transformers of all sizes were fastened to specially installed anchors;
- Support bearings of the turbine equipment were replaced by those seismically qualified.

All equipment rated up to four balls, produced by the industry in those years, including pumps, heat exchangers, the 6 and 0.4 kV sections, valves, relays, control and instrumentation devices was tested on a three-dimensional seismic bench at an acceleration of 2-3 g in Ivangorod, near Leningrad. Supplementary measures were taken to increase its seismic resistance and strength. Listing them all is impossible.

Note that, after the fire at the Armenian NPP in 1982, a completely new electricity supply system was installed (DAR). After the Chernobyl accident, a number of measures were taken at the Armenian NPP to enhance safety, particularly to improve the reliability of cooling water supply for safety related loads in the reactor facility.

At the same time, a program was elaborated to modernize the plant in accordance with OPB-82 (General Safety Regulations) requirements, which call for three safety systems. Plant modernization based on OPB-82 appeared to be economically unfavourable, so in August 1988, it was decided "not to modernize the plant but to put into operation the additional capacities at the Razdan State Regional Power Plant and to shut down the units at the Armenian Nuclear Power Plant". An earthquake in the city of

Spitak on 7 December 1988 precipitated this process and the NPP units were shutdown: Unit 1 on 22 February 1989 and Unit 2 on 18 March 1989.

In 1993, the Government of Armenia adopted a decision to recommission the Armenian NPP: in the first instance to start up Unit 2, since part of the primary SG-5 header of Unit 1 had been cut out for research purposes. It was decided to start the unit at a safety level exceeding the design level by providing additional measures to upgrade the safety and seismic stability of buildings, equipment, machinery and safety systems.

The measures were divided into stages. In the first stage, before start-up of the unit, a large number of measures were taken; the main ones were as follows:

- The number of accumulator batteries per unit was increased from 1 to 2, and two separate, independent auxiliary power supply channels of category I were established, at the same time the number of reversible motor-generators were increased from 2 to 4. New seismically qualified batteries manufactured by the VARTA Company replaced the accumulator batteries.
- The number of diesel-generators was increased from 3 to 4, and two separate, independent auxiliary power supply channels of category II were established.

As a result of a comprehensive investigation of the sprinkler system, technical and organizational measures were implemented making it possible to achieve a fully functional system without replacement of the pumps:

- Expulsors were installed in the steam generator headers;
- The hot injection pipe (200-mm diameter) into the pressurizer was dismantled;
- The tightness of the steam generator and the reactor coolant pump rooms was improved using a special sealing material, based on technology from the American company, Promatec;
- Two additional 4-metre level gauges were installed in each steam generator for supplementary protection and interlocking based on the level in the steam generator;
- A vast amount of work was done optimizing the protection and interlocking of the nuclear steam supply system, keys for removing protections and interlocks were reduced to a minimum, and the industrial seismic protection system and diesel generator system start-up circuits were simplified;
- The steam generator level controllers were replaced by Fisher company products;
- All work to strengthen the load bearing structures of the main building, the diesel generator station, electrical equipment racks, the component cooling water pumping station, etc., was conducted in accordance with design documentation developed by the General Designer after the Spitak earthquake in 1988;
- Equipment made by the Siemens company was installed for diagnosis of the primary circuit equipment and Diameter > 100 mm piping.

After the restart of Unit 2, the implementation of safety upgrading measures was resumed, with the main ones as follows:

- Emergency system of gas extraction from the steam generator headers was installed;
- Check valves were installed on emergency injection lines to the primary circuit;
- Replacement of seismically non-qualified relays, detectors and secondary gauges with those seismically qualified;
- The relief valves of the loops and sealing tubes of the main stop valves were dismantled;
- All station-wide safety systems supplied from two units were switched to supply from Unit 2 alone;
- A large volume of work was performed on construction of two systems to supply safety-related loads with component cooling water involving two independent spray ponds. Unfortunately, the work was not completed and the systems remain under construction.

In 1998-1999 we plan to carry out the following:

- Install 7 MSIVs on the pipelines from SG and on the main steam header;
- Replace the pressurizer safety valves to solve the problem of the reactor vessel protection from the cold over-pressurization;
- Replace steam generator safety valves;
- Put the spray ponds into operation;
- Install a diesel pump to feed the steam generator during beyond the design basis accidents;
- Put the diagnostic system into operation; and
- Build an autonomous sealing system for the shaft of the main circulating pump GTsN-317, thereby freeing two emergency feed pumps.

Our intentions aimed at the safety enhancement of Unit 2 are given in detail in the "Programme for safety, reliability and operational culture upgrading activities of Unit 2 of the Armenian NPP for the period 1997-2000" document.

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SOFTWARE IN SAFETY APPLICATIONS

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INTRODUCTION

The use of software in nuclear power plants is increasing. AECL has pioneered the use of computers for control with the use of digital control computers at Douglas Point (1966). AECL continued this leadership with the use of computers in safety systems with the introduction of PDCs (Programmable Digital Comparators) for the CANDU 6 stations in the early 1980s.

Nonetheless, there are a number of issues across the work-wide software industry related to use of software in general and to safety applications in particular. There is no agreed upon measurable definition of acceptability for the engineering of safety critical software within the technical community. There are no widely accepted practices for the specification, design, verification and testing of safety critical software. These issues are further compounded since software reliability cannot be achieved through testing; i.e. unlike hardware systems, exhaustive testing of all possible outcomes is impossible.

AECL's success can be attributed in part to the establishment of some fundamental principles in the approach to software design for safety applications. First, is the principle of simplicity. That is, the functionality of the safety critical portion of the system must be limited to only what is essential to carry out the safety action. Non-safety functions should be displaced to other auxiliary systems to the extent practical. Secondly, safety design must be done on an overall system basis, not just on the computers or software; e.g. redundancy, fail-safe states, diversity, etc must be built in at the system design level.

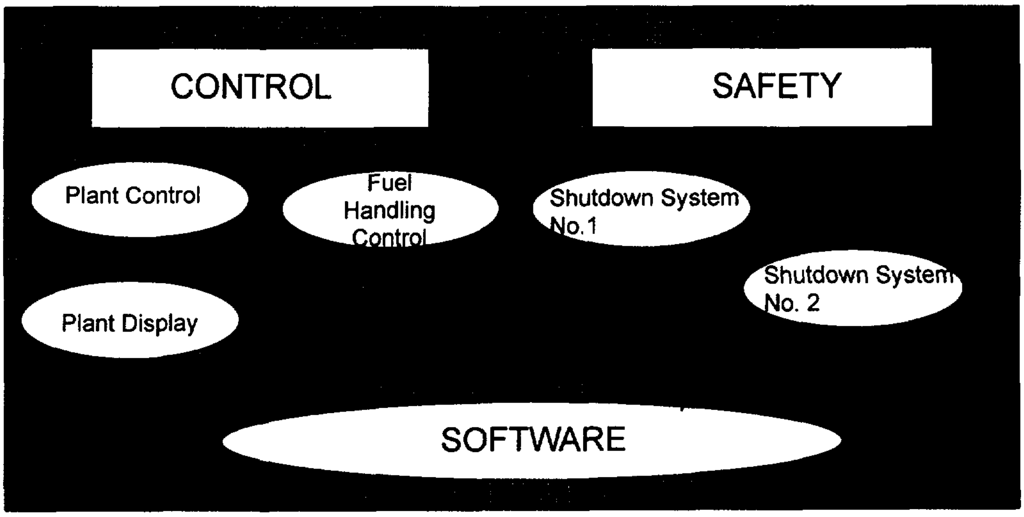

FIG. 1. Increasing use of software in CANDU NPPs.

AECL APPROACH FOR SAFETY CRITICAL SOFTWARE

In addition to these fundamental principles, AECL has developed a rigorous approach to the software development of safety critical software. The key elements of the CANDU Safety Critical Software Lifecycle (see Figure 2) are summarised below:

1. "Formal" (mathematical) specification of requirements: Documentation must be prepared to describe clearly the required behaviour of the software using mathematical functions written in a notation that has a well-defined syntax and semantics.
2. Review and Verification: The outputs from each development process must be reviewed to identify that they comply with the requirements specified in the inputs to that process. In particular, those outputs written using mathematical functions must be systematically verified against the inputs using mathematical verification techniques.
3. Reliability Demonstration Testing: Reliability of the safety critical software must be demonstrated using statistically valid, trajectory-based random testing.
4. Validation Testing: The executable code (integrated with the target hardware) must be tested to demonstrate conformance to the system level requirements.
5. Independence: Independence of design and verification staff shall be maintained to ensure an unbiased verification process.

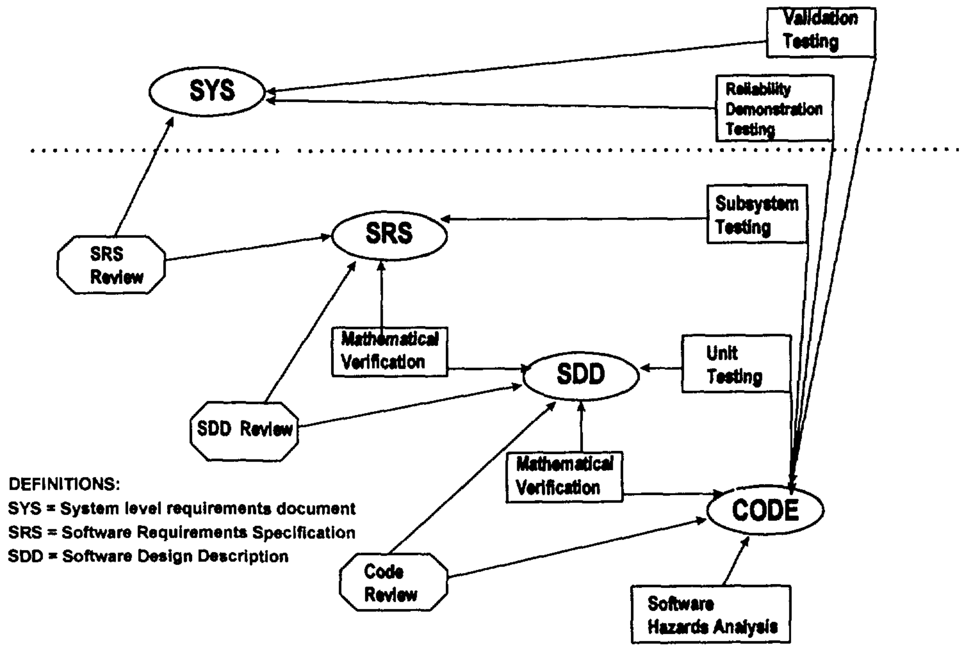

FIG. 2. CANDU safety critical software development lifecycle.

CANDU SAFETY CRITICAL SOFTWARE DEVELOPMENT METHODOLOGIES

AECL has developed two distinct software development methodologies that utilise the above mentioned lifecycle. They are the Rational Design Process (RDP) and the Integrated Approach (IA).

The RDP can be characterised as a methodology where the required behaviour of the software is defined using mathematical functions written in a notation which has a well defined syntax and semantics. The input/output behaviour is defined in tabular format. The IA uses a graphical functional notation to specify the functional software requirements. Since the notation of the software design is mathematical, executable code is generated directly from it without the need for a manual coding step.

AECL has utilised these methodologies successfully for several projects, namely Wolsong 2, 3, 4 for both the IA (Shutdown System No. 1, SDS1) and the RDP (Shutdown System No. 2, SDS2). This work was done using a team of AECL and KOPEC staff. These methodologies are also being used for the Qinshan 1, 2 project for SDS1 and SDS2.

RESULTS

One obvious measure of the quality of the methodologies used is to assess the results of the various testing stages of the lifecycle. The results for Wolsong 2,3,4 PDCs are summarised below.

Results from Wolsong 2, 3, 4 SDS1 & SDS2 Software Testing

Shutdown System	Unit & Subsystem testing	Validation Testing	Reliability Testing
SDS1	0	0	0
SDS2	0	0	0

There were zero errors from all the phases of testing. This clearly demonstrates the validity and the benefits of the approach taken by AECL for safety critical software.

CONCLUSIONS

The results from the projects already completed using the AECL methodologies and tools demonstrate that the approach to CANDU software design is at the fore-front of world-wide software technology. The software development methods and verification and validation (V&V) techniques are among the most rigorous and best in use in the nuclear industry today.

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SUMMARY OF PC SIMULATION USE FOR CANDU I&C APPLICATIONS

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Abstract

This paper will identify some of the low fidelity, PC based simulations previously used for CANDU Nuclear Power Plant (NPP) Instrumentation and Control (I&C) systems applications for training, commissioning and design purposes. This technique is suitable for:

- control strategy and performance confirmations,
- man-machine interface design capabilities,
- applications requiring a relatively short development and implementation period
- those applications requiring the capability for on-going simulation development or progression in a user-friendly manner without allowing the provision of expert simulation staff.

1. INTRODUCTION

The standalone PC based simulation approach has been successfully used to assist training, commissioning and design instrumentation and control activities for various CANDU projects over the past two decades. The general evaluation criteria that had been developed for selection and approval of such simulation solutions required that the problem:

- be relatively complex in nature
- have an attendant high degree of difficulty for successful task completion,
- perhaps be performed infrequently ,and
- have a significant consequential penalty for incorrect or incomplete task resolution.

2. TRAINING

The Computer Assisted Learning techniques have been developed to aid in the training of unit first operators, control technicians and technical staff for CANDU nuclear generating stations. Standalone PC based training simulations have been found to be an effective training alternative at sites prior to full-scope simulator availability (or upgrade implementation) when used in conjunction with both classroom and on-the-job training programs. The PC based standalone simulations have been particularly effective at developing the supporting knowledge and skills necessary to operate an individual system or device to provide a strong ***building block*** approach to integrated unit operations understanding and mastery.

The benefits in the long-term arise from the improved efficiency of delivering station systems orientation, advancement and refresher training as well as additional training support for licensing qualification examinations. ***Reductions in training costs*** can be expected from reduced trainee costs (i.e. fewer hours spent away from the job, few failures and reduced repeat attendance), reduced trainer costs (i.e. lower work load which replaces routine presentation with specific problem resolution, computerized reviews, tests and progress reports) and a general reduction in training facility costs (i.e. equipment can be used in the plant on all shifts reducing the need for central training classrooms).

3. COMMISSIONING

The use of PC based simulation has proven to be a very effective tool to assess and solve high priority plant I&C commissioning problems on a fast turnaround basis. The use of simulation for the resolution of such problems allows for a very detailed control related study to be completed with the development of correspondingly detailed corrective recommendations. There is high confidence placed on findings obtained from the simulation study allowing faster executive reviews and approvals of the technical proposals and action plans.

The results from the simulation assessment can be used to develop specific commissioning actions and procedures; and to provide specific design guidance and input for the final design solution. The simulation can be used repeatedly to assess the original unstable, cyclic process boiler level control performance in order to be able to accurately diagnose the situation and identify the necessary corrections. This approach minimizes the number of stress events the actual unit will be

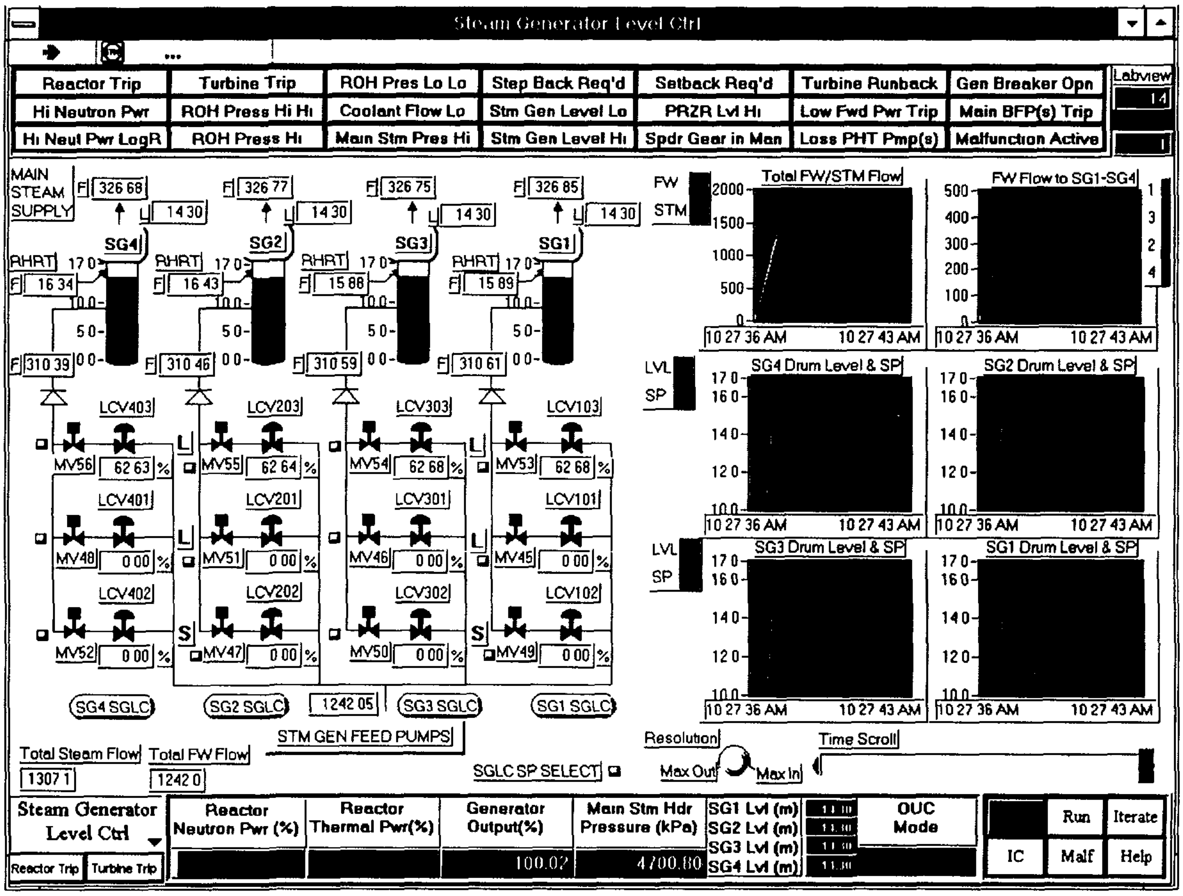

FIG. 1. Typical PC based simulator steam generator level control display.

subjected to and so minimize the potential risk involved. The same simulation assessment methodology could be applied to the deaerator level and pressure control problems over a longer time frame, as that problem may not have the same immediacy, but with very similar positive corrective results being successfully identified and implemented.

4. DESIGN PROCESS

Integration of simulation into the design process provides a design team with the means to conduct ***rapid prototyping*** of concepts, to assess alternate layout and control strategies, to test and exercise control logic over operation state changes and to review human system interface information sets for completeness and correctness.

Having a simulation platform which provides the capability to select and customize simulation modules from a library means that designers, expert in the plant design, can configure simulation models to address the design task at hand (i.e. review, assess, testing, verification, validation,etc). The ***resource load on the project is reduced*** since design experts prepare the simulation code with less knowledge of thesimulation technique (e.g. use simulation as a tool) as opposed to simulation experts preparing the code with less knowledge of the station.

The advantages to the project are a ***higher quality final design product*** with assured functionality, fewer omissions, fewer opportunities for human error and more complete and correct operator information presentation sets.

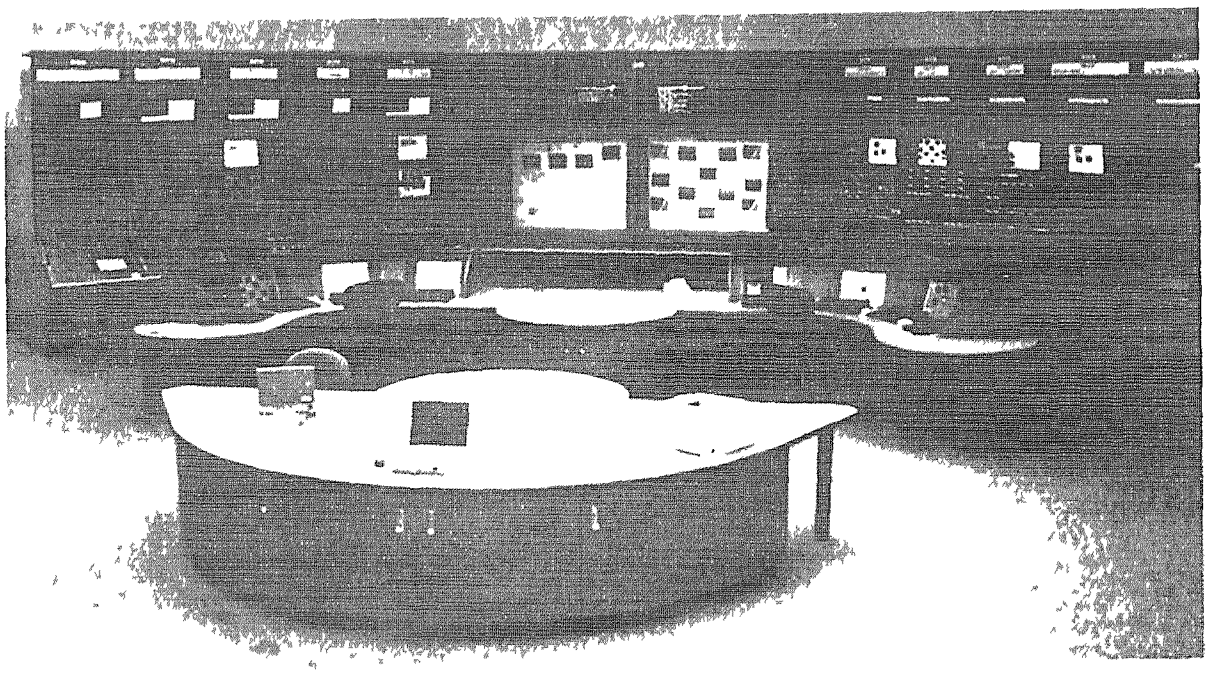

FIG 2 CANDU 9 control centre mockup

5. FUTURE OPPORTUNITIES

The advantages seen from these rather modest attempts to integrate simulation use into training, commissioning and design activities points the general way for future simulation based engineering activities. The complete integration of simulation into all phases of NPP life cycle development and support with one simulation platform should allow significant improvements in quality and product with attendant decreases in time required and associated costs. In particular, the provision of a simulation tool that uses ***one model set*** for engineering development, safety analysis and training applications should be the most significant contributor for these goals.

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XA0053608

INHERENT SAFETY FEATURES OF THE CANDU REACTOR

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CANDU® reactors offer a high degree of defence-in-depth to protect against upsets and accidents. A significant part of this defence-in-depth relies on inherent features of the CANDU system or features that have been implemented as part of the system for a long time and can therefore be regarded as proven. Some aspects of the design meet the defined requirements for passive systems. Noteworthy features are mentioned below.

The CANDU reactor possesses a number of intrinsic safety features owing to its particular design characteristics. Firstly, CANDU reactors are based on a pressure-tube design rather than a pressure-vessel design. Thus, the pressurized portion of the reactor is limited to the fuel channels, which are of relatively small diameter. This helps to ensure that containment integrity and sufficient fuel cooling are not jeopardized following a pressure-boundary failure. Secondly, since components in the primary heat transport system are located above the reactor core, thermosyphoning of the primary coolant is possible in the event that pump power is lost. That is, the CANDU reactor can passively remove decay heat. Thirdly, the fuel channels are immersed in a cool, low-pressure heavy-water moderator, which is circulated through heat exchangers by pumps. The moderator serves as an emergency heat sink capable of removing decay heat if delivery of emergency core coolant fails. As a result, the likelihood of severe core damage is inherently low. In addition, the moderator conditions facilitate access to the reactor for reactivity control devices, detectors, and shutdown systems.

There are certain inherent safety features of the CANDU reactor that are attributable to the reactor physics. The reactivity state of the core is relatively constant with time due to the use of on-power refuelling and the fact that reactivity coefficients are all small in magnitude. Reactivity fluctuations can be easily controlled; the excess reactivity available from the fuel is small and the relatively long lifetime of prompt neutrons in the reactor precludes rapid changes in power levels. Furthermore, the fuel-channel lattice is optimized for maximum reactivity. Hence, an event that results in relocation of fuel causes a reduction in reactivity.

The CANDU reactor is equipped with two, independent, redundant, and diverse shutdown systems. Each shutdown system is fully capable and is not dependent on the reactor regulating system. The shutdown systems are fail-safe and passive. The first shutdown system consists of shutoff rods which are vertically oriented above the reactor core and are gravity-driven with spring assistance. The second shutdown system uses liquid poison injection into the moderator. This system is horizontally configured, and the liquid poison injection is accomplished using fast-acting valves between a high-pressure helium tank and the poison tanks. The probability of a power transient without reactor shutdown has been shown to be extremely small.

The CANDU reactor features various systems to safely mitigate a loss-of-coolant accident (LOCA). Emergency core coolant (ECC) can be injected into the primary heat transport system. The ECC is passively supplied from water tanks connected to a high-pressure gas tank; additional ECC supply is available from an elevated water reservoir referred to as the dousing tank. The dousing tank also provides containment cooling and depressurization through a gravity-driven water spray system that condenses

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escaped steam. Hydrogen levels in containment are controlled with the aid of igniters or passive recombiners. As mentioned previously, the moderator acts as an emergency heat sink if ECC injection fails, thereby preventing severe core damage. Moreover, the tank containing the moderator (known as the calandria vessel), is itself surrounded by a large water-filled shield tank. Consequently, if there was a LOCA coincident with a loss of ECC and moderator cooling (i.e. a triple failure), a severe core damage accident would progress very slowly. Moderator heatup and boiloff would require several hours. The water in the shield tank could retain debris inside the calandria and hence prevent melt-through of the fuel to the containment internal structures, for a period of about one day. This would allow a long time for accident management, restoration of emergency systems, and public emergency plan execution. Lastly, the CANDU system design is such that the containment building would at most be subjected to a modest challenge, even for severe core damage events. The configuration of the fuel and the core pressure boundary inherently precludes direct containment heating or high-pressure melt ejection, which could fail a containment structure.

XA0053609

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THE NUCLEAR PLANT ANALYZER ATLAS

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An overview of the plant analyzer ATLAS [1] is given describing its configuration, the process models and the supplementary modules which enhance the function of ATLAS for a range of applications in the field of reactor safety analyses.

ATLAS is designed according to client / server principles. The data server takes a central position; it administers the resulting data to simulation models and keeps them ready for all systems linked to the network. These systems includes the visualization and interactive communication (MONITOR) and supplementary systems such as decision support, diagnostic and surveillance. The server stores updated data as well as the history of all arriving data. Simulation, server, visualization programs and supplementary programs are run as separate computing processes (clients) which communicate via data interfaces. The processes may be distributed to one or more computers within a network. The system actually runs under UNIX, and hardware of different manufacturers can be used.

ATLAS is applied in the field of NPP safety analyses both at GRS and external organizations (e. g. technical authorities, universities, etc.).

The model basis for the simulator is provided by best-estimate codes for the thermal fluid dynamics in the reactor coolant loops, and for the physical processes in the containment. The thermal fluid dynamics is modeled by the ATHLET system code [2]. ATHLET includes a software environment for the simulation of the instrumentation and control systems of the plant. For descriptions of the processes within the containment, the GRS-codes RALOC and COCOSYS are employed; for the simulation of severe accidents the third-party integral code MELCOR.

On the basis of ATLAS several detailed analysis simulators with a large library of pictures have been developed at GRS. The library includes graphics to assist the analyst in understanding the plant behavior, graphics available at the plant control room (Siemens PRISCA system), mimics of control and auxiliary systems for operator interaction and several pre-defined trend sets with important process values.

Qualified input decks have been produced for NPP specific analysis simulators. These analysis simulators enable the analyst to quickly investigate system behavior for essential accident paths detected in safety and risk analyses. Up to now, four analysis simulators were finished for the pressurized water reactors (PWR) of Broldorf and Neckar (GKN-2) and the boiling water reactors (BWR) of Gundremmingen and Krümmel. The simulators for Unterweser (PWR) and Philippsburg I(BWR) are under construction.

An analysis simulator for the NPP Balakowo, a VVER-1000/320 reactor is being developed. This work is being performed by GRS and the Russian partner VNIIAES and OKB Gidropress.

Application of the various simulators on measured operational transients demonstrate that the process model ATHLET and the interactive features of ATLAS can realistically simulate BWR and PWR transient plant behavior. With these simulators, plant safety can be evaluated, measures for the prevention of severe accidents can be investigated and improved and plant personnel training in the field of thermal-hydraulic accident behavior can be complemented.

Applications of ATLAS to a variety of tasks, the essence of a multifunctional tool, are made possible by coupling additional modules to the simulation data. Such modules may help the analyst by decision support, implement operating and emergency procedures, diagnose the state of the plant, or couple the simulation models to plant data to obtain tracking capabilities.

The Reliability Advisory System (RELADS) supplies ATLAS users with information from Probabilistic Safety Assessment, complementing "deterministic" simulation information. RELADS, an example of a decision support system, provides answers to questions concerning the most probable failure or the most dangerous failure, to occur in the simulated plant situation.

To analyze Emergency Operating Procedures (EOPs), a module based on the on-line expert tool G2 has been created and coupled to the data server. A knowledge base was generated, stating general knowledge about procedural steps, their actions and the sequence in which they should operate. Each procedural step has been derived from a general class "procedure step", leading to a limited number of building blocks. These objects, together with event handling and logic, may be interactively connected to produce the desired procedure. An interface to the data server has been written by utilization the G2 Standard Interface.

Future developments will concentrate on the issue of tracking. For the safety analyst, tools to set up simulation corresponding to a measured plant state will increase the efficiency of this work. Other areas such as diagnosis, control system and man-machine design may well profit from this effort. The computing platform will be extended to include Windows NT.

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IAEA-SM-353/16P

OPTIMAL ELEVATION OF SEISMIC SUPPORT FOR CEDMs CONSIDERING DYNAMIC INTERACTIONS WITH RV OF A PRESSURIZED WATER REACTOR

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The optimal elevation of the seismic supports for the control element drive mechanisms (CEDMs) of a pressurized water reactor is determined by considering the dynamic interactions between the motions of the reactor vessel and the CEDMs. Fig.1 shows a schematic view of the unsupported CEDMs and the reactor vessel. Various support concepts can be investigated to have the system perform proper functions and retain structural integrity during seismic events. The dynamic interactions between the reactor vessel and the CEDMs are investigated with a mathematical model that idealizes the reactor vessel as a single mass primary system and the CEDMs as multiple secondary systems whose masses are different each other, as shown in Fig.2. The complex frequency response functions of the primary system and the secondary system are expressed, respectively, as;

$$H(\omega)_{P,d/a} = \frac{-(1+\mu_t) - \omega^2 \sum_{j=1}^{n} \frac{\mu_j}{\omega_j^2 - \omega^2 + 2i\varsigma_j\omega\omega_j}}{\omega_p^2 - (1+\mu_t)\omega^2 + 2i\varsigma_p\omega\omega_p - \omega^4 \sum_{j=1}^{n} \frac{\mu_j}{\omega_j^2 - \omega^2 + 2i\varsigma_j\omega\omega_j}} \qquad (1)$$

$$H(\omega)_{S,j,d/a} = -\frac{\omega_p^2 + 2i\varsigma_p\omega\omega_p + \omega^4\left\{\mu_j \sum_{k=1}^{n} \frac{1}{\omega_k^2 - \omega^2 + 2i\varsigma_k\omega\omega_k} - \sum_{k=1}^{n} \frac{\mu_k}{\omega_k^2 - \omega^2 + 2i\varsigma_k\omega\omega_k}\right\}}{\left\{\omega_p^2 - (1+\mu_t)\omega^2 + 2i\varsigma_p\omega\omega_p - \omega^4 \sum_{k=1}^{n} \frac{\mu_k}{\omega_k^2 - \omega^2 + 2i\varsigma_k\omega\omega_k}\right\}\left(\omega_j^2 - \omega^2 + 2i\varsigma_j\omega\omega_j\right)} \qquad (2)$$

The mean square displacement responses of the system to the stationary random base excitation with constant power spectral density S_0 can be obtained from the following equations;

$$E\left[x_p^2\right] = S_0 \int_{-\infty}^{\infty} \left|H_p(\omega)\right|^2 d\omega \qquad (3)$$

$$E\left[x_j^2\right] = S_0 \int_{-\infty}^{\infty} \left|H_{S,j}(\omega)\right|^2 d\omega \qquad (4)$$

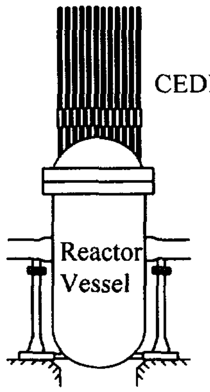

FIG. 1. Descriptive view of the reactor vessel and the free standing CEDMs of a pressurized water reactor.

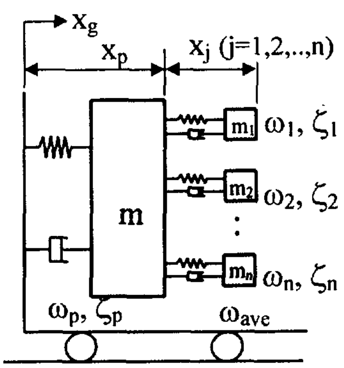

FIG. 2. Simplified mathematical model representing the reactor vessel and the CEDMs.

By defining the tuning frequency ratio f (= ω_{ave}/ω_p) and the frequency bandwidth of the secondary system β (= $(\omega_n-\omega_1)/\omega_p$), the responses of the system can be calculated as a function of f and β for a given mass ratio, the number of secondary systems and damping ratios. The optimal tuning frequency ratio f^{opt} and the optimal frequency bandwidth of the secondary systems β^{opt} are determined to minimize the response of the primary system [1]. If the seismic supports are added to the CEDMs as shown in Fig.3, the modal characteristics will be changed resulting in the change of dynamic interaction effect between the reactor vessel and the CEDMs. One of the dominant factors modifying the modal properties of the system is the support elevation h. Fig.4 shows the variation of f and β with the variation of the support elevation, h/l, which is obtained by performing modal analysis of the supported CEDMs with the reactor vessel uncoupled. It can be seen from Fig.4 that f^{opt} and β^{opt} (0.95 and 0.36, respectively, for μ_t=0.045, ζ_s=0.02 and n>15, from Ref.[1]) can be achieved at h/l= 0.83. The seismic responses are calculated to verify that the optimal support location determined from the idealized mathematical model subject to the random base excitation is applicable to the actual structural system, i.e., the reactor vessel and the CEDMs excited by seismic disturbances. The seismic response analysis for the coupled model of the reactor vessel and the supported CEDMs is performed using ANSYS 5.1 computer code[2]. Fig.5 compares the maximum seismic acceleration responses of the reactor vessel and the CEDMs for the variation of the support elevation. It can be confirmed from Fig.5 that the seismic acceleration response of the reactor vessel is minimized at the optimal elevation of the CEDM seismic support.

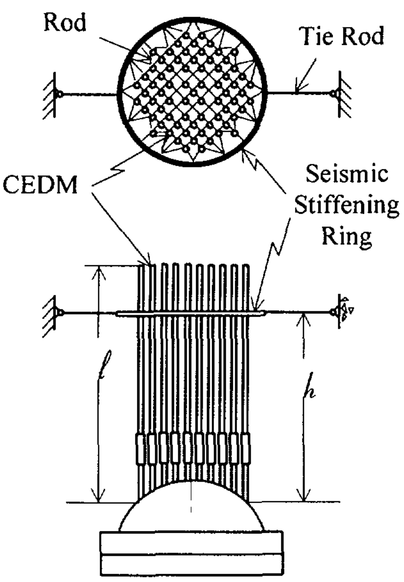

FIG. 3. Conceptual configuration of the seismic supports for the CEDMs.

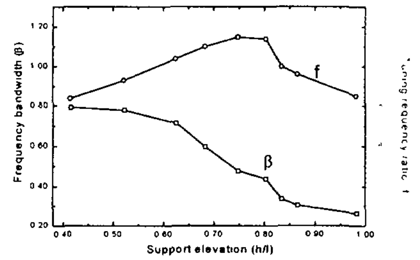

FIG. 4. Variation of tuning frequency ratio f and frequency bandwidth β with the variation of support elevation.

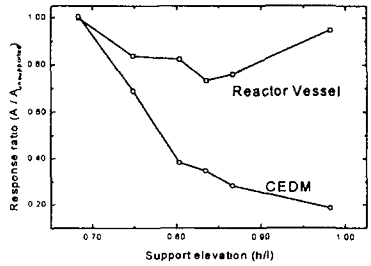

FIG. 5. Variation of the maximum seismic acceleration responses of the reactor vessel and the CEDMs with the variation of the support elevation.

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IAEA-SM-353/17P XA0053611

A DESIGN APPROACH TO ADDRESS INTERSYSTEM LOCA

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An intersystem loss of coolant accident (ISLOCA) is defined as a class of events in which a break occurs outside the containment in a system connected to the RCS, causing a loss of reactor coolant inventory. The loss of inventory might disable emergency core cooling systems, and create a radionuclide release pathway that bypasses containment. Although the ISLOCA risk is a relatively small contributor to the total probability of core damage, the scenario influences risk perceptions because containment is bypassed at the outset. The Korean Next Generation Reactor (KNGR) is designed to reduce the level of ISLOCA challenges to all systems interfacing with the RCS. The general design features to address ISLOCA challenges consist of an increase of the system design pressure and incorporation of design features that terminate and/or limit the event by isolation or pressure relief. Although these design features provide protection against ISLOCA challenges, they could cause adverse effects such as: 1) to incur additional risks as a result of reducing ISLOCA risk, 2) to increase equipment costs, 3) to reduce operation flexibility, and 4) to make the system design more complicated. Without establishing a compliance method properly, the ISLOCA requirements might be conservatively applied and be unduly excessive. Therefore a design approach to address ISLOCA for the KNGR is established deliberately and judiciously.

The compliance methods for ISLOCA go beyond the traditional approach for assuring the ability of systems that interface with the RCS to withstand an overpressurization event. It is treated as a beyond-design base event. Therefore, the ASME code rules are not applicable and the associated standard review plan guidelines are not considered for the design of piping system. The KNGR low-pressure intersystems are designed to withstand the consequences of ISLOCA challenges practically and reasonably. The design approach is shown in Fig. 1. The aim of this design approach is to reduce

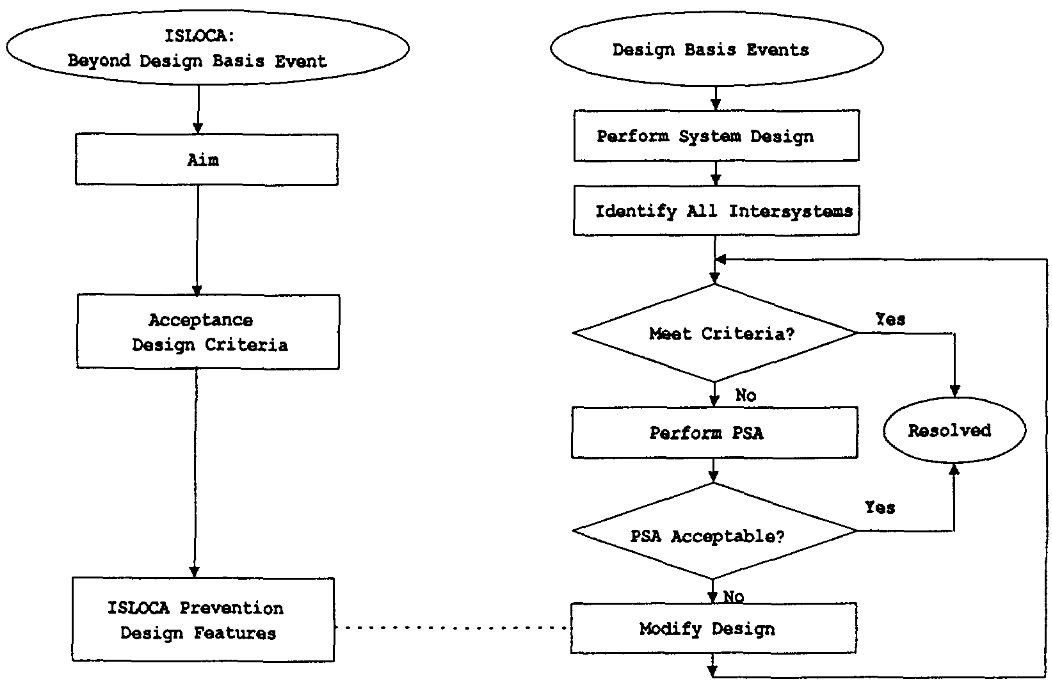

Fig. 1. KNGR design approach against ISLOCA.

core damage frequency (CDF) and public risk due to overpressurization and failure of low-pressure systems. Important factors considered to accomplish this aim are:

- the likelihood of pressurizing initiators such as human error and Pressure Isolation Valve (PIV) failures;
- component fragility; and
- consequence of coolant loss and radionuclide release during ISLOCA, and post-ISLOCA recovery operations.

A method to comply with these factors is to follow a rigorous inspection program with improved surveillance, maintenance, and test procedures after construction. However, this paper focuses on compliance methods applicable to the design process. To ensure satisfaction of ISLOCA criteria, design guidance for the KNGR has been presented. The preliminary system design reflects only traditional design base events at the beginning stage of the design process. The system design is then evaluated for ISLOCA challenges. Both deterministic and probabilistic approaches are applied for the evaluation. The acceptance criteria are addressed based on the ISLOCA requirements as described in the Korean Utility Requirements Document [1].

Systems susceptible to ISLOCA should be designed such that the following acceptance criteria are satisfied:

- The system retains its structural integrity throughout the event (structural integrity is preserved if, by definition, the system maintains its pressure boundary despite distortion and/or loss of function).
- Any leakage caused by the event is limited to the makeup system capabilities and offsite doses are limited to a small fraction of those specified in 10 CFR 100.

For piping systems normally open to the atmosphere and for certain large tanks and heat exchangers, it would be difficult or prohibitively expensive to design such systems for full RCS pressure. For the cases where the first criterion is not met, a system design to meet the second criterion only is acceptable.

Operating experience has indicated that, in three cases where the interface boundary failures occurred in U.S.A., there was a degradation of the PIVs due to personnel errors. The historical data support a PSA approach. In the design process, PSA insights are used to select an appropriate design among design options, strengthen the design against previously known vulnerabilities, characterize the design, and evaluate the design for ISLOCA. A probabilistic evaluation is performed for the cases that do not satisfy the acceptance criteria. The pressurization pathways from the RCS are probabilistically evaluated, considering human and hardware contributions to ISLOCA for the interconnecting systems. The failure frequency data used are per Korean URD [1] and NUREG/CR-5102 [2]. The mean ISLOCA CDF is estimated to be 2.0E-6/reactor-year in the PWR ISLOCA assessments [3, 4, 5]. By comparison with this, the estimated probability of 1.0E-12/reactor-year is selected as an acceptance criterion because the contribution to core damage is considered too small. For those systems meeting the probability criterion, the system is considered acceptable. If the ISLOCA vulnerability is significant, system design modifications are made. The design pressure of the low pressure systems is increased to a high-pressure rating of 40 percent of RCS normal operating pressure, that is, 6.31 MPa, so that the ultimate rupture strength is at least equal to the RCS pressure, or design features are incorporated that eliminate the potential for ISLOCA events, or terminate and/or limit their scope.

All NSSS systems or subsystems have been reviewed to identify all the possible pressurization pathways located outside containment and having direct interface with the RCS during some modes of operation, but are not designed to the high pressure rating of 6.31 MPa.

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DEVELOPMENT OF DIGITAL SAFETY SYSTEM, CORE PROTECTION CALCULATOR SYSTEM CPCS) USING NETWORK AND COMMERCIAL OS

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The CPCS is a safety system calculating the DNBR (Departure from Nucleate Boiling Ratio) and LPD(Local Power Density) trip signals in a four redundant channel configuration. The CPCS reads the CEA (Control Element Assembly) position signal from two redundant sensors. This resulted in one out-of–two-logic for CEA related events. This two channel redundant signal is a weakness of the system, giving some spurious trip signals from the CEA sensor itself, signal isolator, or other component malfunctions

The new CPCS will utilize the open computer system to get more support from various vendors and more resistance to equipment obsolescence problems [1]. The network technology is used widely across the system. The system consists of four redundant channels; each consists of CPC (Core Protection Calculator), CEAC (CEA Calculator), Gateway, CEA MUX (CEA Multiplexer), and OM (Operator's Module). The equipment in a single channel is tied together using the channel network (CH-NET). The data flow within a channel is unidirectional from CEA MUX to CEAC and CPC, from CEAC to CPC. The OM displays the status of each component and calculated output. Gateway provides an interface to other system, and provides the electrical and functional isolation between safety system and non-safety system. Figure 1 shows the configuration of the system. The use of network technology makes sharing data easy and this reduces a lot of components within the system, compared to the old system. In the old system, a number of isolation devices have been used to isolate the channel fault between the interconnected channels or between safety system and control system. The fiber optic network communication provides the required electrical and functional isolation. Also the network protocol can support built-in reliability of dual redundant configuration.

With reduced components and ease of data sharing, the new CPCS can also check additional data integrity. The system utilizes the CEA MUX to share two redundant CEA position signals and one set of contact signals among four channels of CEAC. These contact signals were used only for CEA control systems. To utilize these signals in both safety system and control system, an enormous number of isolator devices would have been needed in the system. This requirement for isolation along with separation was one of the reasons so many sensors were used in nuclear power plants. With the use of network, signals can be shared easily among redundant channels or between safety and control systems.

For safety systems, the simple and deterministic nature of the system is emphasized. In this respect, the system architecture is very important. The software architecture is priority-driven pre-emptive scheduling, with only timer interrupt allowed. For the software, the design process itself is important and requires that the process be manifested by design documents. The use of formal specification language may help show the coherence and completeness of the software functional requirements [2]. The application software should be developed by applying rigorous mathematical methods. Currently we are evaluating the formal methods in developing the software requirement specifications. Our observation is that each method has its own characteristics and we are considering applying different methods for different aspects.

The operating system should utilize a commercial real time operating system that has enough experience. Most of the commercial real time operating systems have the scalability, making OS with only necessary parts and supporting priority-driven pre-emptive scheduling. Qualification of the commercial operating system should be ascertained through EPRI dedication guidelines that recommend support from the vendors and testing software at specific application [3].

Digital safety system design using a network makes data sharing easier and allows simpler system configuration. Use of commercial OS and formal methods in software development increases system reliability and reduces development time. A reliable digital safety system will contribute to the plant safety and availability.

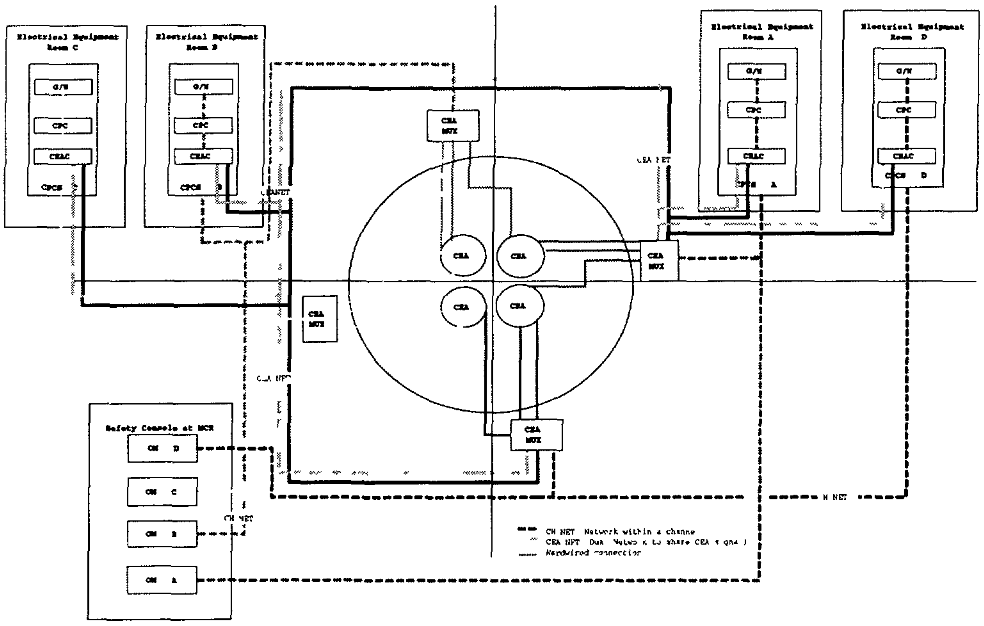

Fig. 1. CPCS system configuration with network interfaces within system.

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XA0053613

IAEA-SM-353-19P

AN EXPERIMENTAL ANALYSIS OF PASSIVE HYDRAULIC VALVE CHARACTERISTICS

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A kind of three-way check valve, a so-called hydraulic valve was proposed as a substitute for the density lock of passive reactors such as SPWR (System-Integrated Pressurized Water Reactor). The valve functions to separate the borated water from the main coolant loop for normal operation, and prevent the insurge of the water into the loop for shutdown, and to remove decay power when the main coolant flow rate is insufficient.

The two operational modes can be explained by the motivating principles of the hydraulic valve as follows. In normal operation (when the head of the main coolant pump is sufficient), the forced flow of coolant in the primary system causes pressure to drop across the piston, raising the piston to close the natural circulation path. On the other hand, in abnormal operation (when the head of main coolant pump is insufficient), the decreased differential pressure drop causes the piston to fall and opens a natural circulation loop. Thus, the opened natural circulation path between the core and the water pool causes borated water flow into the core, the reactor is shutdown and the residual heat is removed from the core.

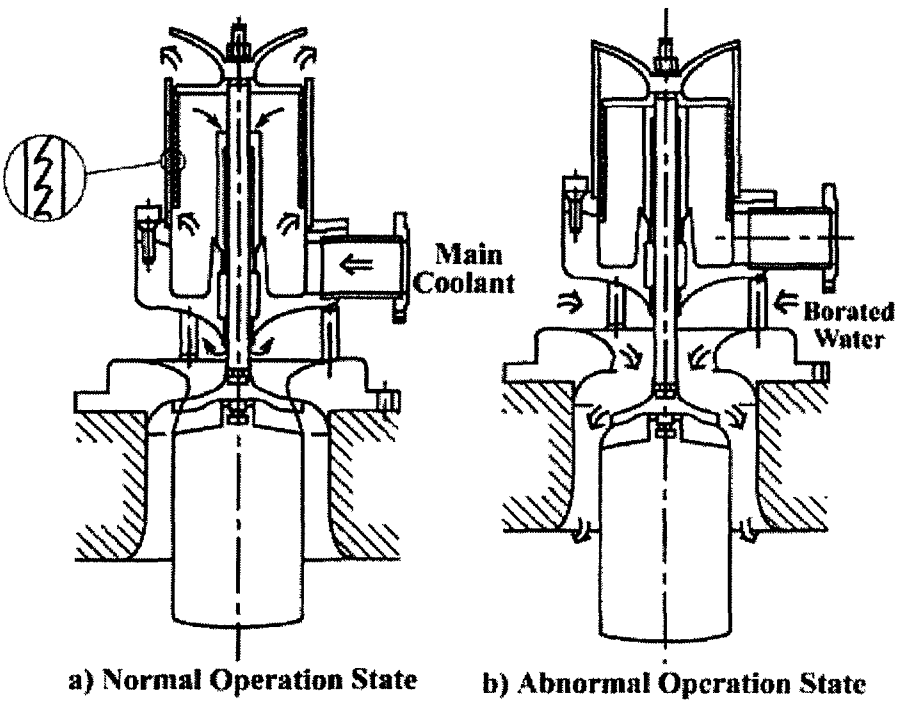
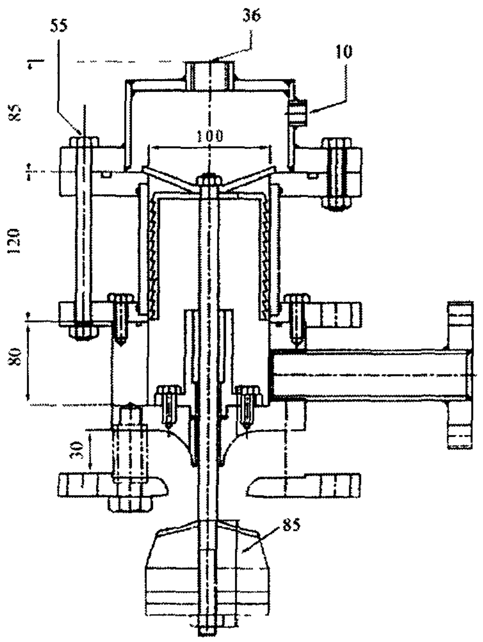


Fig. 1. Operation mechanism of the hydraulic valve. a) normal operation state flow rate above 40 percent) b) abnormal operation state (flow rate below 40 percent)

A 1/3-scale experimental model was set up to prove the operability and to analyze the valve characteristics (Figs. 2, 3). The model valve was manufactured such that the similitude of the prototype valve (SPWR) is observed. Main parameters of the prototype and the model valve are shown in Table 1.

TABLE 1. DESIGN PARAMETERS OF MODEL AND PROTOTYPE VALVE.

	Flow Rate (%, kg/sec)		Velocity(m/sec)	ΔP(kPa)	Lifting Force(kg)
Prototype Valve	40	13.2	3.97	25.2	170.0
	100	36.1	10.8	200.0	1260.0
Model Valve	40	0.88	1.73	5.82	5.0
	100	2.02	3.958	34.95	26.0



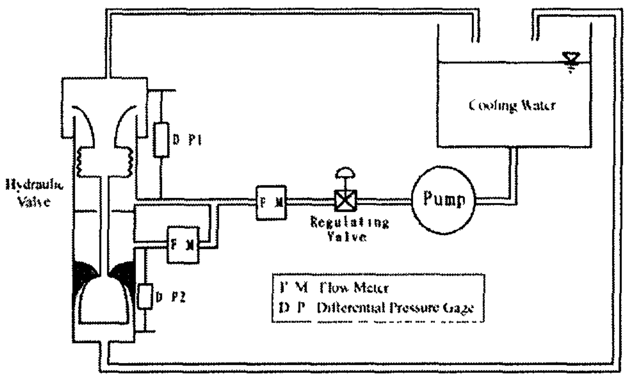


Fig. 3 A scheme of experimental rig.

Fig. 2 A Dimension of hydraulic valve.

Experimental results show satisfactory operability of the valve. The valve casing and opening was distinct at the set flow rate, which is varied according to the balance weight. The valve characteristics, that is, the differential pressure drop across the piston, were correlated as a function of angle, number and pitch-size of the teeth. The pressure drop increased with the number of teeth, but the pressure drop decreased if the pitch was greater or less than a certain value (8.24mm). Therefore, there is an optimum ratio of pitch tooth to suitable pitch size. The pressure drop coefficient (K) defined by diffuser model was modified as follows to fit the experimental results.

$\Delta P = K\ 1/2\ \rho\ u^2$, $K = 1.1K_0(N)^{-0.26}(A)^{-0.78}$, N is number of teeth, A is angle of teeth, K_0 is theoretical pressure drop coefficient of typical diffuser.

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XA0053614

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A FEASIBILITY STUDY ON THE LOAD FOLLOW OPERATION WITHOUT BORON ADJUSTMENTS IN 1300 MWe KNGR

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Applicability of the MSHIM control technology, which had been developed by Westinghouse and characterized by elimination of the adjustment to the boron concentration during load maneuvering, to the preliminary designed 1300MWe KNGR(Korean Next Generation Reactor) Cycle-1 was examined by simulation based on one-dimensional transient analyses under the conditions of relatively large xenon worth and high amplitude of xenon oscillation. These are well known factors obstructing the smooth control of the core reactivity and the axial power distribution during power maneuvering.

The original control bank configurations of KNGR were rearranged to have a suitable rod worth and a capability of power distribution control for the MSHIM control strategy. The existing two PSCEAs and three regulating banks were replaced by independently operable groups of control banks composed of four M banks and one AO bank prepared for reactivity and power shape control, respectively. All control rods consisted of B_4C and shutdown banks were preserved. One of the M banks, M0 bank, was fully inserted in the core during base load operation for the compensation of xenon buildup in part load operation. The rod worth and overlap size of each bank are shown in Table 1, which was determined by repeated simulation of load follow operations with various bank worth and overlap size. As shown in Table 1, the AO bank has the relatively heavy worth so as to maintain the monotonic relationship between AO(Axial Offset) and AO bank motion.

The axial power distribution is controlled to keep the axial offset as constant as possible during the power maneuvering. It is performed by repositioning the AO bank with respect to the difference between target axial offset(TAO) and AO at that time. The TAO value in load maneuver was settled by approximately 10~15% more negative value than that in base load operation in order to provide more effective axial offset control in both negative and positive directions. This procedure was accomplished in intermediate period, referred to as transition day, between base load operation and load follow [1].

Three types of power maneuvering, daily maneuvering, extended weekend load follow, and spinning reserve capacity were simulated to evaluate the capability of the selected load follow technology. All simulations were performed in two limiting burnup states, i.e., at 3% BOL and 90% EOL. Xenon buildup effect during part load operation becomes more significant as time goes by BOL, and the amplitude of axial xenon oscillation becomes lager as time goes by EOL.

TABLE 1. CONTROL ROD PARAMETERS FOR THE MSHIM CONTROL STRATEGY

Bank	Rod Worth at Near BOL [pcm]	Rod Worth at Near EOL [pcm]	Overlap [%]
M0	492	663	0
M0+M1	887	1032	33
M0+M1+M2	1506	1695	33
AO	1295	1300	

Total peaking factors were evaluated by synthesis procedure between radial peaking factors and axial power distributions for all types of load follow pattern in order to ensure the compliance of safety limits in the viewpoint of nuclear design. The summarized results for all cases are listed in Table 2. Except for the case of weekend maneuver at near EOL, all results are held below the design limit on power distribution (2.5~2.6 in total peaking factor, 2.69 for the Yonggwang Unit 3&4) of conventional CE type reactors [2]. If the gray rod can be utilized and the modification of control rod configuration is allowed in assembly, it could be also overcome.

In this work, CASMO-3/MASTER[3,4] code system and ONED94[5] were used to analyze the static and transient core behaviors, respectively.

By showing that the calculated total peaking factors are maintained below safety limit, it is proved that load follow operation without boron adjustment is feasible with simple control rod motion only.

TABLE 2. SUMMARY OF LOAD FOLLOW OPERATION

Burnup	Maneuver	TAO_{LF}	Maximum F_z	Maximum F_Q
Near BOL	Daily	TAO_{BASE}-10%	1.419	2.1264
	Weekend	TAO_{BASE}-15%	1.504	2.3343
	Spinning Reserve	TAO_{BASE}-10%	1.421	2.1163
Near EOL	Daily	TAO_{BASE}-12%	1.618	2.1514
	Weekend	TAO_{BASE}-15%	1.824	2.5326
	Spinning Reserve	TAO_{BASE}-12%	1.693	2.1505

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XA0053615

IAEA-SM-353/23P

A DESIGN APPROACH TO COMPUTERIZED PROCEDURE SYSTEM FOR KOREAN NEXT GENERATION REACTORS

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The KNGR (Korea Next Generation Reactor) represents an effort to develop a standard design for an advanced nuclear power plant with significant safety improvement and cost reduction. For KNGR, the concept of an advanced control station is applied to the design of the main control room (MCR). Computerized compact workstations and digital operating systems provide operators with electronically generated images and information on the status of the plant systems.

The CPS (Computerized Procedure System) is a computerized, instead of paper, procedure to be imbedded into the workstations of the KNGR MCR. The basic functions and features of COPMA-II, the second version of a computerized procedure developed at the OECD Halden Reactor Project, were adopted in the development of CPS. As in COPMA-II, the functions in CPS include: (a) procedure editing (i.e., procedure writing, installation into the CPS framework, and modification), (b) procedure search (i.e., access to a specific procedure), (c) real-time access to the plant state, (d) indirect component control, (e) automatic procedure flowchart generation, and (f) place peeking of procedure execution. Since COPMA-II is a laboratory research product, instead of a commercial product developed for industry use, many improvements are needed to be useful for KNGR operation. The design approach for the KNGR CPS was based on a review of the problems with conventional paper procedures and man-machine interface concepts, to be developed for the computerized workstations of the KNGR MCR.

This paper describes the current design of the prototype CPS based on COPMA-II as well as projected features provided in the first version of CPS, for future development. The principal functions and features of the prototype CPS are described in detail and potential advantages over conventional paper procedures as well as over COPMA-II, are highlighted. Since the design process is important in the development of a complex software system like a computerized procedure, the design process plan as well as features of CPS, are also described. Since usability and suitability are critical in a computerized operating procedure, there is a brief description of human effort in the design process of CPS. Since a computerized procedure is intended for actual plant operation, and should satisfy requirements in the KNGR URD (Utility Requirement Document), some of those requirements and ways CPS would meet them are also described.

The design approaches presented in this paper are still being refined; additional review and tests are due in the future. The current design of CPS (See Figure 1) will be evaluated for suitability and effectiveness in operation using a variety of test facilities, such as a partial dynamic mockup, a full dynamic mockup with KSNP (Korea Standard Nuclear Plant) models, and finally using a KNGR full scope simulator. KEPCO (Korea Electric Power Corporation) has had limited experience in the development and application of computerized procedures to nuclear plant operations. Therefore, efforts have been made to reflect other experiences such as those of the French with the N4 and OECD Halden Reactor Project (HRP), where extensive laboratory studies on computerized procedures (the COPMA series) were done.

CPS is expected to drastically decrease the navigation workload, reduce time for procedure manipulation and execution and increase the convenience of operation by providing context sensitive task information with less effort. However, some challenges in designing an effective computerized procedure are decreases in operator competence and vigilance during operation by reliance on the computerized procedure. It should be clear that design of such a complex and critical system as CPS should carefully consider human factors. Integration with other computerized systems in the KNGR MCR is also a key issue in designing the KNGR CPS. Several countermeasures are considered for mitigating possible problems associated with a computerized procedure. Further discussion and research results on development of the KNGR CPS will be available in the future.

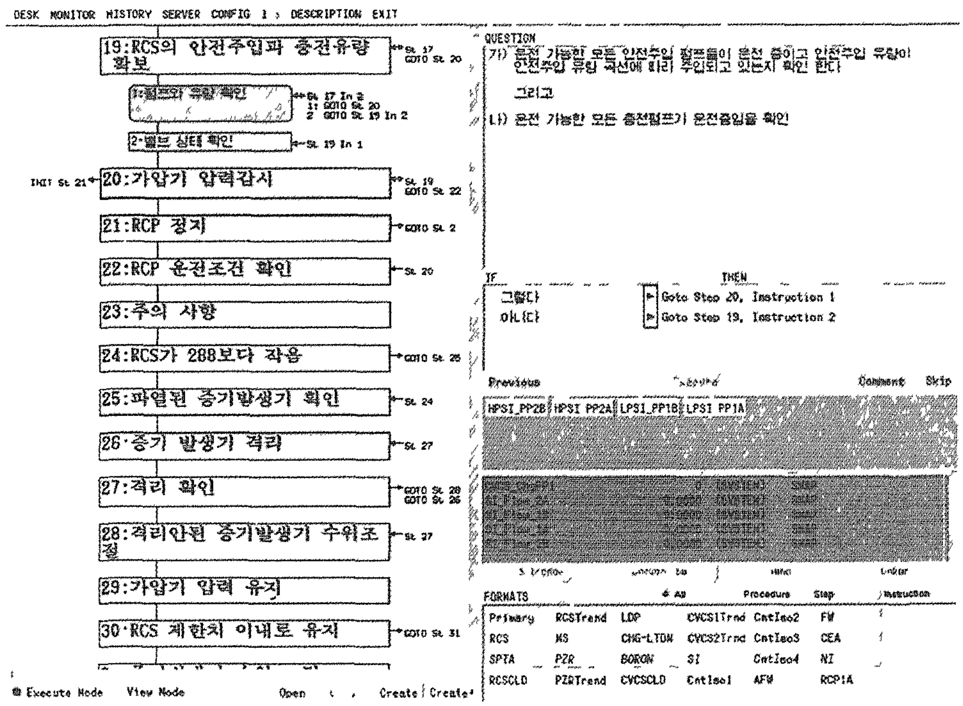

FIG. 1. A snapshot of the KNGR CPS

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XA0053616

DEVELOPMENT OF THE ADVANCED I&C: ASICS AND ADIOS

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Digital technology was recently introduced to instrumentation and control (I&C) systems in nuclear power plants. It then actively proceeded through the industry so that intelligent technologies could be applied to the operation and operator support system. In this paper, the Automatic Startup Intelligent Control System (ASICS) and Alarm and Diagnosis-Integrated Operator Support System (ADIOS) which are connected to the functional test facility (FTF), are described.

ASICS automatically controls the PWR plant from cold shutdown to 5% reactor power. The ASICS has a supervisor system and a distributed control system. The supervisor system has a program to control the distributed control system and a knowledge base designed from general operating procedures and operator experience. The supervisor system is implemented using an intelligent real-time expert system shell G2. The distributed control system has four automation modes: the Heating I mode, Heating II mode, Critical mode and Secondary mode. Each mode has controllers and keep-up bands. The keep-up bands have a check-up function to start each automation mode or to hold on some process variables to allow operator action. The ASICS function is verified to receive control-input signals from the FTF and send control results to the FTF. Through this method, the developed control algorithm is evaluated in a real-time environment.

For the performance test of Mode I, the heat-up rate was set to 27°C/hr, the start temperature and the target temperature were 60°C and 176.6°C, respectively. The pressure set point was $24kg/cm^2$. From the Mode I test, the heat-up time required to reach the target temperature was 7 hours. The automatic temperature controller saved about 2 hours compared to manual operation for heat-up. The measured heat-up slope was uniform and the pressure was controlled constantly. For the operation test of Mode II, the heat-up rate was the same as in Mode I, the target temperature was 292°C. The pressure set point was $157kg/cm^2$. From the Mode II test, the heat-up time required to reach the target temperature was 8 hours. The automatic temperature controller saved about 3 hours in heat-up time compared to manual operation. The measured P-T curve was located in the required operation boundary. From these tests we could confirm that the supervisory control rules and the controllers of the distributed control system were well designed, as expected. ASICS could save about 4 hours in heat-up time and reduce the chance of repeated operations and trips. ASICS could also reduce the operator's burden .

The overall objective of ADIOS is to improve the operational performance of the man-machine interface system by integrating alarms, process values and diagnostic information to an expert system focused on alarm processing. The ADIOS was implemented using the G2 real-time expert system shell. Its knowledge base is constructed on the process knowledge of a Korean nuclear power plant, using some advanced alarm processing concepts. Every alarm is treated as an object of an alarm class. Objects consist of various attributes, including tile message, process value, alarm firing set-point, activation status, priority, acknowledgement or reset status, causal alarm, level precursor, and so on. Therefore, each alarm object with those attributes contains all the necessary information for processing and display in the system. Some of the attributes of an alarm object change their values dynamically during a run of ADIOS. The values of process variables and status are fed into the attribute, "process value," of the corresponding alarm objects. The "acknowledgement or reset status" controls the flashing display depending on the acknowledgement status of the alarm when it is activated or deactivated. The attributes, "casual alarm" and "level precursor" prioritize the alarm on the basis of the relationship with other alarms. The alarms in ADIOS initially had their own default priorities which are then changed according to plant-mode dependency, equipment-status dependency, multi-setpoint relationship, or some other method. The dynamic prioritization is determined by using "Rules" and "Procedures" of the expert system shell. The plant-mode dependency is a method through which the alarms activated as a consequence of the plant mode change are de-prioritized from the default priority. Multiple set-point relationships use the relationships among several alarms on the same process parameter. For example, when both the low and low-low level alarms of a steam generator are activated, the priority of the low alarm should be lowered. The activated alarms are displayed as rectangles around the process value on the process overview mimic and chronological alarm lists with different colors in according to their priorities. The diagnostic function module will be incorporated into ADIOS to estimate and inform the operators of the root causes of some complicated failure behavior. To verify only the functional effectiveness of the alarm and diagnosis system without investigating operational performance, it was tested and operated with the test scenarios generated in the real-time test facility by activating malfunctions to simulate abnormal plant conditions.

In applying the digital technology to nuclear power plants, the safety and reliability of the plant should be ensured. Testing and validating the function and performance of a digital system should be conducted in a realistic environment prior to installation in an NPP. The objective of instrumentation and control FTF is to test and validate new digital control and protection algorithms, the alarm reduction algorithm, and the performance of operator support systems, etc. The FTF provides a simulated testing environment as an experimental test bed. The FTF software consists of a mathematical model simulating a three-loop, 993 MWe pressurized water reactor, and a supervisory program that comprises the instructions necessary to run the FTF. The hardware equipment provides an interface between host computer and simple test panel or developed target systems to be tested. The interface module can provide an Ethernet or VXI interface to the prototype using shared memory and the display page for the value of simulated variables. The graphic user-interface supports an easy and friendly interface between FTF and users. It is implemented through a Picasso-3 graphic tool developed by the Halden Reactor Project. The FTF is applied to an ASICS and ADIOS as shown in Figure 1, to test its algorithm and performance. The results of the test show good operational performance of the FTF in normal and transient conditions.

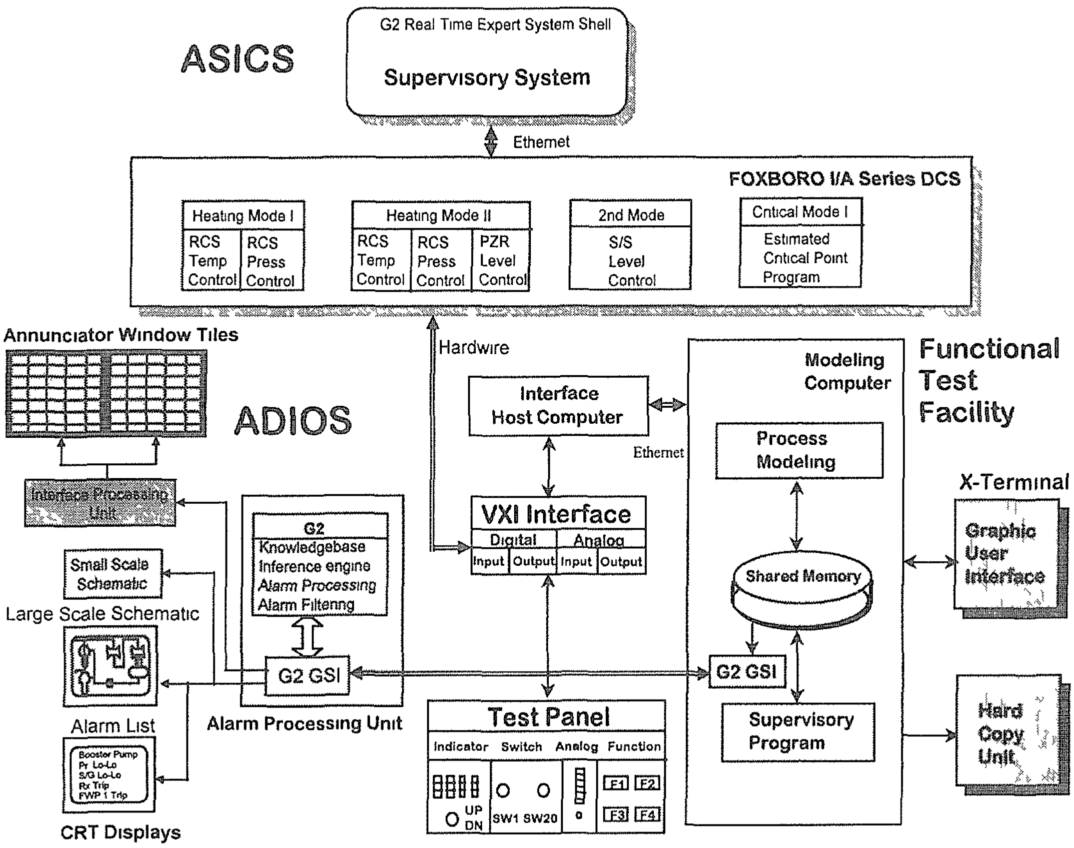

Figure 1. ASICS and ADIOS with test facility

ACKNOWLEDGEMENT

This work was conducted under the nuclear research and development program of the Ministry of Science and Technology in Korea.

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XA0053617

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IN-VESSEL RETENTION AGAINST WATER REACTOR CORE MELTING ACCIDENTS

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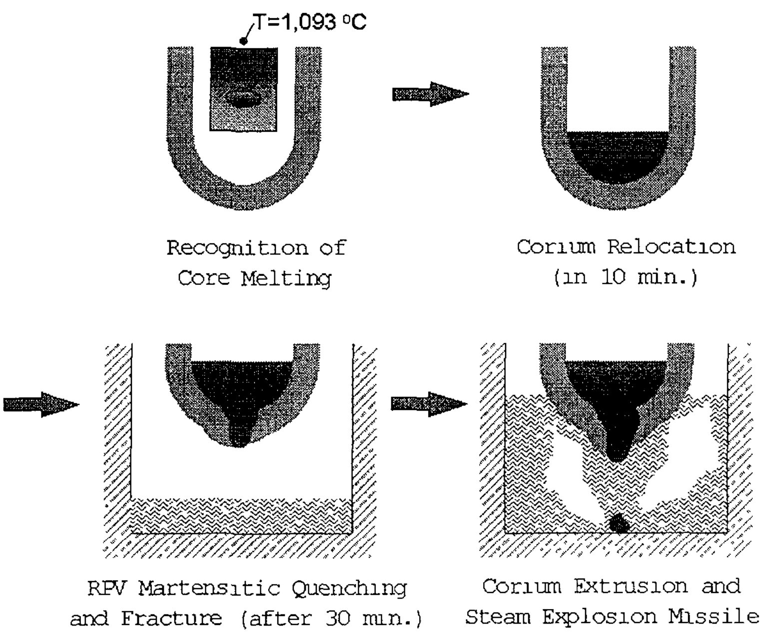
Following the TMI-2 accident, advanced water reactor designs have been developed with an emphasis on severe accident management capability. Among these are designs incorporating ex-vessel corium cooling and/or in-vessel corium retention capabilities. The ex-vessel management approach, however, appears to involve several shortcomings. For example, corium ejected through the failed RPV lower head tends to agglomerate, compromising the long-term coolability. In addition, issues associated with the containment direct heating and the steam explosion are yet to be resolved. For this reason, more recent designs prefer the in-vessel retention of corium by maintaining the structural integrity of the reactor vessel lower head.

Several advanced water reactor designs incorporate provisions to manage the escaped core debris outside the vessel. The ex-vessel management approach involves the difficult tasks of corium cooling and steam explosion control. If one can secure the integrity of the lower head by maintaining coolability in the region, core-melting accidents will be arrested within the vessel at a significant advantage over the ex-vessel measures.

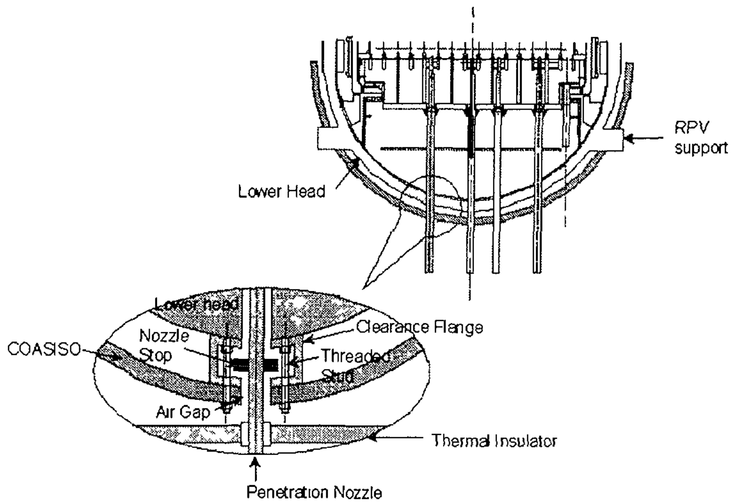
The in-vessel retention can be achieved by full flooding* of the reactor cavity to cool the external wall of the lower head, thereby avoiding structural failure by creep rupture. Application of this approach to large power reactors is not trivial because of relatively short time between the detection of core melting and the lower head failure. Therefore, special design features to facilitate rapid flooding are essential to the success of in-vessel retention. We have examined the thermal and structural behavior of a large power reactor during a severe accident to evaluate the full cavity



* Predicted lower head behavior for slow flooding scenario

flooding approach. Based on the result, an improved in-vessel retention method using gap structures is proposed for rapid cooling of the lower head.

This paper examines potential options of in-vessel retention for large PWRs as a function of time spent to wet the external wall of the lower head. Important options were evaluated by detailed thermohydraulic and mechanical analysis models developed and benchmarked with experimental data. The results of this study elucidate that the risk of lower head failure by either creep rupture or quench fracture increases rapidly with the amount of time spent for external wall wetting. To facilitate rapid wetting without the installation of major pumping systems, gap structures for the lower head are proposed. The external gap structure, designated as COASISO**, is shown to be an effective option for in-vessel retention of both new and operating reactors against severe accidents.



** COASISO design figure

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XA0053618

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APPLICATION OF POOL-TYPE REACTORS RUTA FOR DISTRICT HEATING IN FOSSIL FUEL DEFICIENT REGIONS

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Considering the acute problems of district heating in many fossil fuel deficient regions of Russia, the application of nuclear heat is an important technical and economic objective. Employment of the small pool-type reactor RUTA, under joint development by the RDIPE and the SRC RF-IPPE, could be a practical option. Development of RUTA reactors of 10 to 50 MWt and Nuclear Heating Plants based on RUTA, aims at an autonomous nuclear power source to supply heat for small towns or city districts. Recent studies reveal a considerable market for small nuclear heating plants in the country, especially in the Far East and Northeast regions of Russia [1]. A pool-type reactor RUTA [2] has obvious and transparent safety characteristics based on simplicity and reliability of design, and inherent safety features using the laws of nature.

From the viewpoint of safety, the RUTA reactor has the following special qualities:

- The primary reactor loop is inside a single reactor vessel (a water tank or pool), operating under atmospheric pressure. This eliminates reactor primary circuit rupture accompanied by a rapid coolant loss and uncovering of the reactor core.
- The large bulk of reactor coolant in the reactor pool enables the accumulation of residual heat for longer time periods, up to several days, even when there is a complete malfunction of normal and emergency heat removal systems.
- Low values of reactor parameters: atmospheric pressure at the water surface in the reactor pool, no water boiling in the pool.
- Low power density, approximately 15-kW/ l, in the reactor core.
- An integral primary circuit layout: primary heat exchangers are in the reactor pool.
- Natural circulation of reactor primary coolant in normal and accidental conditions.

Development of a 4-unit NHP with 55 MWt RUTA reactors for the city of Apatity, Murmansk is under way. Feasibility and economic studies reveal a high competitiveness for the project.

One of most important conditions of nuclear project implementation is positive public opinion and public acceptance. To gain it, convincing and evident proof of safety is necessary. The best way to obtain such proof is construction and successful operation of a prototype or demonstration plant. A demonstration prototype is important to confirm the basic design, operation and economic features of the project. To permit more active introduction of the RUTA reactor to the market, i.e., for commercial use in fuel deficient regions of Russia, construction of the NHP RUTA demonstration project at the SRC RF-IPPE is under study.

Design work has shown that when coupling the 30 MW NHP RUTA to an existing heating system, an effective variant of its use can be found with the reactor operating mainly in the base mode.

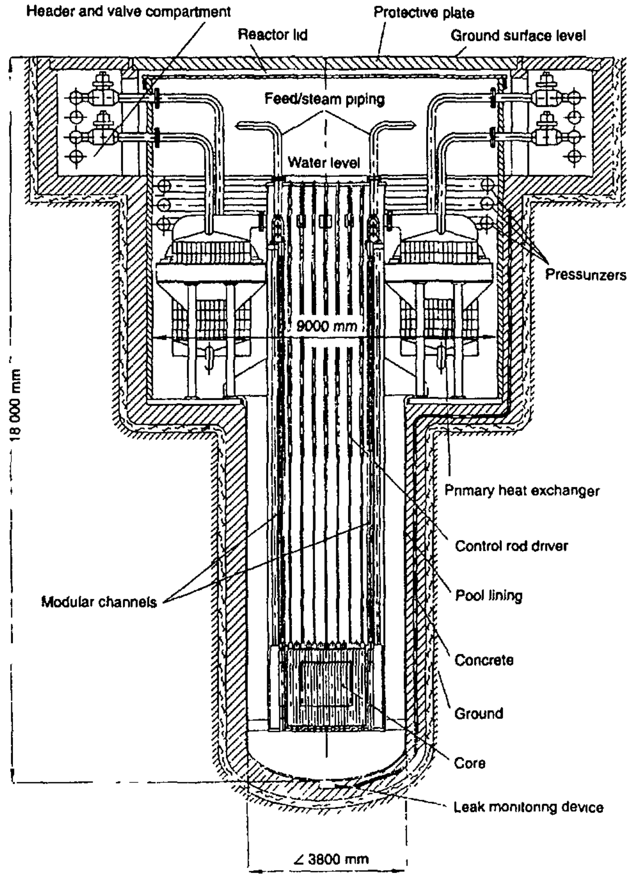

Fig. 1. RUTA-55 reactor section.

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IAEA-SM-353/35P

SAFETY IMPROVEMENT AND RESULTS OF COMMISSIONING OF MOCHOVCE NPP WWER 440/213

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Mochovce NPP is the last of its kind and compared to its predecessors, is characterized by several modifications contributing to the improvement of safety. In addition, it is based on Nuclear Regulatory Authority requirements and on the following documents:

- IAEA - Safety Issues and their ranking for NPP WWER 440/213;
- IAEA - Safety Improvement of Mochovce NPP Project Review Mission; and
- Riskaudit - Evaluation of the Mochovce NPP Safety Improvements (additional safety measures were implemented before commissioning).

The consortium EUCOM (FRAMATOME - SIEMENS), SKODA Praha ENERGOPROJEKT Praha, Russian organizations and VUJE Trnava Nuclear Power Plants Research Institute were selected for design and implementation of the safety measures. They consist of 87 safety issues elaborated in the framework of 10 design areas and 11 issues oriented to operational areas. Safety categories of individual safety issues are defined and ranked according to the approach in IAEA documents.

Implementation of the Mochovce Safety Enhancement Project was divided into three principal phases:

- studies and analyses,
- design preparation phase, and
- implementation of modifications of system or civil engineering structures.

The licensing for the operation of Units 1 and 2 is based on a pre-operational safety analysis report (POSAR) and conducted according to internationally recognized standards. The whole concept evolved from the US NRC RG 1.70; for the accident analyses constituting part of this report, the IAEA document " Guidelines for Accident Analysis for WWER Nuclear Power Plants" was used, taking into account the Slovak legislation in force.

Main milestones of commissioning of Unit 1:

1. April 27, 1998 - first fuel loading
2. June 9, 1998 - first criticality
3. July 4, 1998 - first connection to the grid
4. August 28, 1998 - first nominal power

IAEA-SM-353/36P

THE ALPHA PROGRAMS AND THERMAL-HYDRAULIC FACILITIES AT PSI

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A number of passive Advanced Light Water Reactor (ALWR) designs have been proposed worldwide. These are today in different R&D and design phases. The ALPHA project was initiated in 1991 at the Paul Scherrer Institute (PSI) in Switzerland. The work is directed to the experimental and analytical investigation of two main containment design aspects for the next-generation of "passive" ALWRs: long-term decay heat removal and fission product retention; this poster addresses only the decay heat removal work, describes the facilities, and summarizes the scope and accomplishments of the project in relation to several proposed passive ALWR designs.

All passive ALWRs make use of large passive systems for the transfer of decay heat, following an assumed depressurization of the primary system, from the containment building to either evaporating water pools or to convectively air-cooled structures. The energy removal from the reactor containment involves the condensation of the steam produced by decay heat. This takes place in the presence of non-condensable gases that were either initially already present in the containment (air or nitrogen) or were added later to its atmosphere by core degradation in case of a hypothetical severe accident (hydrogen). These two types of non-condensables have quite different densities and produce different effects. The efficiency of the condensation process and the distribution of the non-condensable gases in the various containment volumes play a key role in determining overall containment behavior. These condensation and mixing phenomena are of main interest for the ALPHA project.

During the first ALPHA-I phase of the program, the effort concentrated on the General Electric SBWR [1]. Both BWR (ESBWR [2], SWR-1000 [3]) as well as PWR passive designs (in particular designs with double concrete containment, such as the EPP [4]) are the focus of the second, current phase, ALPHA-II.

THE ALPHA TEST FACILITIES

PANDA is a large-scale integral-test facility having a modular structure of cylindrical vessels interconnected by piping. Multidimensional effects are allowed to take place by the division of the main containment compartments (DW and WW) in two. Parametric or sensitivity experiments conducted under well-controlled boundary conditions provide more valuable data for code qualification than experiments where the mixing phenomena are distorted by the scale of the facility or other reasons. The facility has been used to investigate integral system aspects for the SBWR, the ESBWR and the SWR-1000.

The smaller-scale LINX-2 facility has been designed for separate-effects tests to study condensation and mixing phenomena such as direct-contact and wall condensation in the presence of non-condensable gases, condensation on finned tubes, venting of steam-air mixtures, and pool thermal mixing induced by single and two-phase plumes. Experiments are conducted at a fairly large scale and

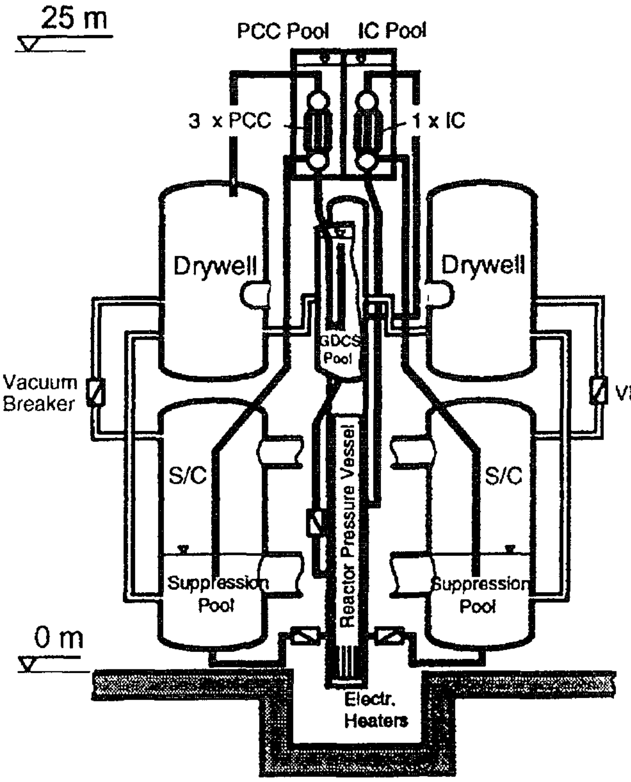

Fig. 1. The PANDA facility.

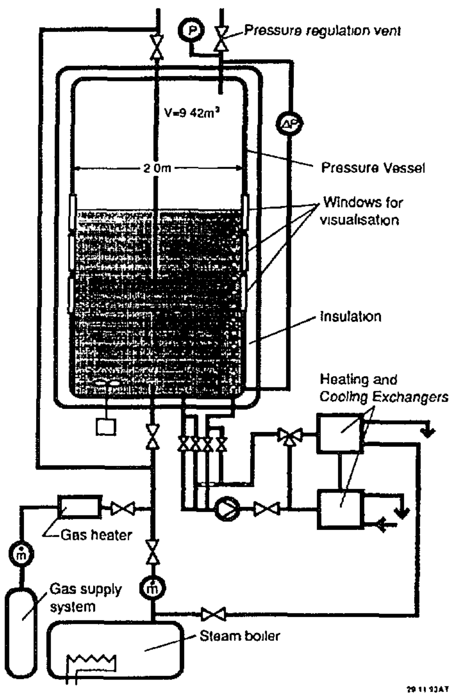

Fig. 2. The LINX-2 facility.

under prototypical pressure and temperature conditions to investigate thermal-hydraulic phenomena of interest to both passive BWR and PWRs.

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XA0053621

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STATUS UPDATE OF TEST PROGRAMMES FOR ADVANCED FUEL DESIGNS FOR ABB-CE AND KSNPP REACTORS

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Introduction

ABB's customers and colleagues have identified their needs for advanced fuel designs including improved fuel reliability, improved fuel cycle economics, higher burnup capability and improved performance. ABB Combustion Engineering (ABB CE) has developed the Turbo™ advanced fuel design to meet these needs. The Turbo design includes: GUARDIAN™ debris-resistant spacer grids, Turbo Zircaloy mixing grids to increase thermal margin and grid-to-rod fretting resistance, value-added fuel pellets to increase fuel loading, advanced cladding to increase burnup, axial blankets to improve neutron economy, and the erbia integral burnable absorber for improving fuel cycle economics (Figure 1). All of these features are directly applicable to the Korea Standard Nuclear Power Plant (KSNPP).

This paper presents the T&H qualification testing done for the 14x14 Turbo design. 14x14 test assemblies containing all advanced features began irradiation in August 1996. Complete qualification tests were also performed for the "I" spring design and other advanced features.

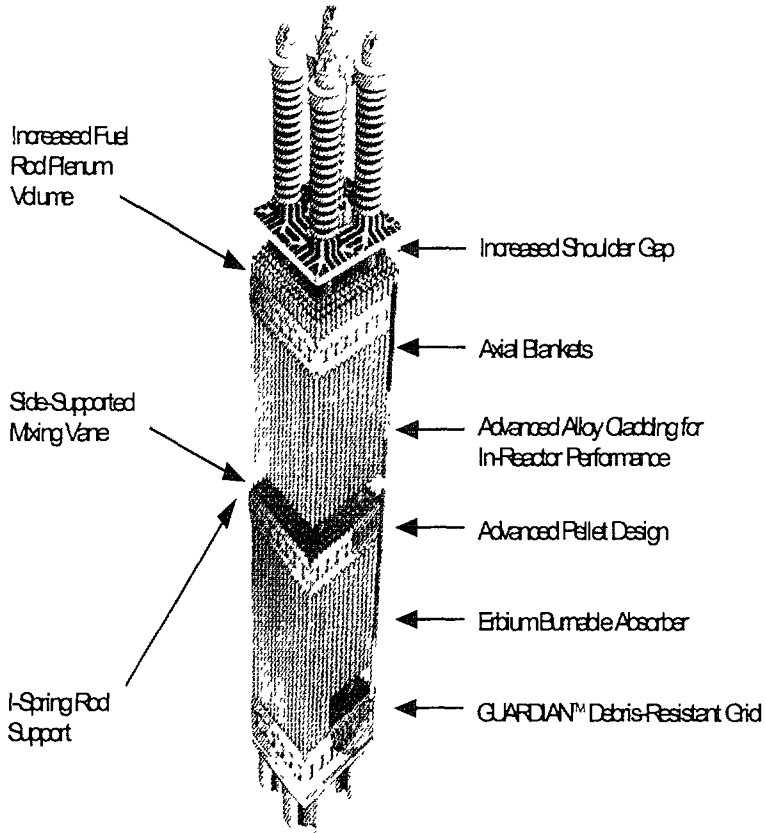

FIG. 1. Turbo fuel.

Assembly Flow Tests

Full scale prototype testing of the advanced fuel design was performed in ABB's test flow facility, TF-2. A single assembly pressure drop test was performed at reactor operating temperature and flow conditions to determine the assembly and component loss coefficients over a range of bare rod Reynolds numbers from 80,000 to 600,000. The test data(Figure 2) demonstrates that the advanced assembly loss coefficient is within design criteria.

A dual assembly endurance test was performed in the TF-2 loop to demonstrate acceptable performance in transition cores. The grid springs were set at the nominal long-term zero gap, the maximum best estimate gap, and the maximum credible gap. A comparison of the measured wear depth versus acceptance criteria (Figure 3) demonstrates excellent fretting wear resistance with the Turbo "I" spring, and acceptable fret resistance with the standard grid. Assembly motion data was also taken and demonstrates no self-excited fuel assembly vibration. In addition, the measured flow in each bundle, based upon the measured component pressure drops was compared to the predicted flow split (Figure 4). This demonstrates good agreement between predicted and measured flow.

Screening Tests and Analysis for Development of Side-Supported Mixing Vanes

In developing the Side-Supported Mixing Vanes several screening test programs were performed. These test programs include overscale air tests to visualize flow patterns downstream of advanced grids, 5x5 and 6x6 cold water loop tests to measure grid pressure drop and velocity profiles with the Laser Doppler Velocimeter and 5x5 Freon Critical Heat Flux (CHF) tests for selecting the final grid design. ABB CE has performed CHF tests for approximately 20 different grid designs over the past 18 years. These screening test programs were performed to understand and measure the flow patterns, pressure drop and velocity profiles downstream of these advanced grids. The results of these tests indicated that mixing vane designs that produced a strong swirl in the subchannel had the greatest impact on DNB performance with the least impact on pressure drop.

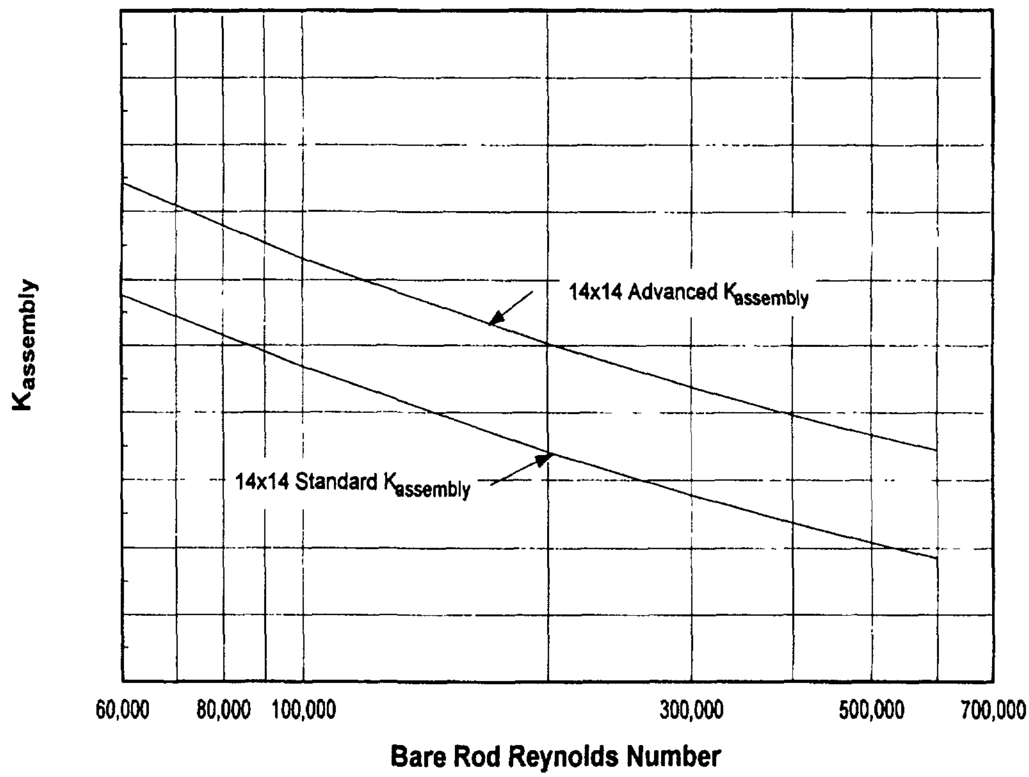

FIG. 2. Fuel assembly pressure loss coefficients.

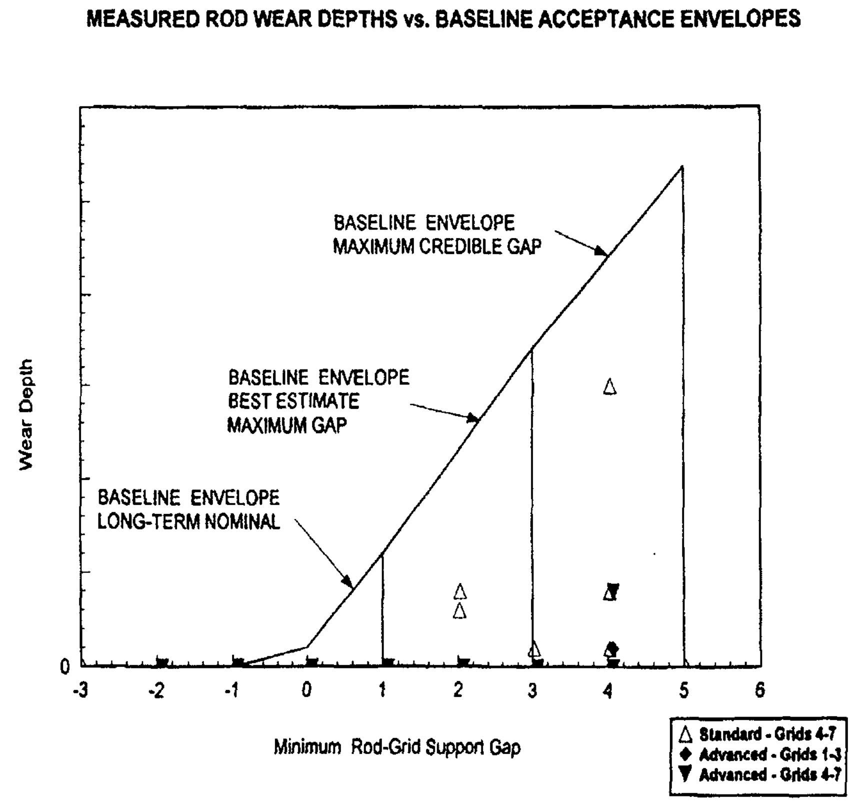

FIG. 3. Comparison of observed wear from endurance test with acceptable envelope.

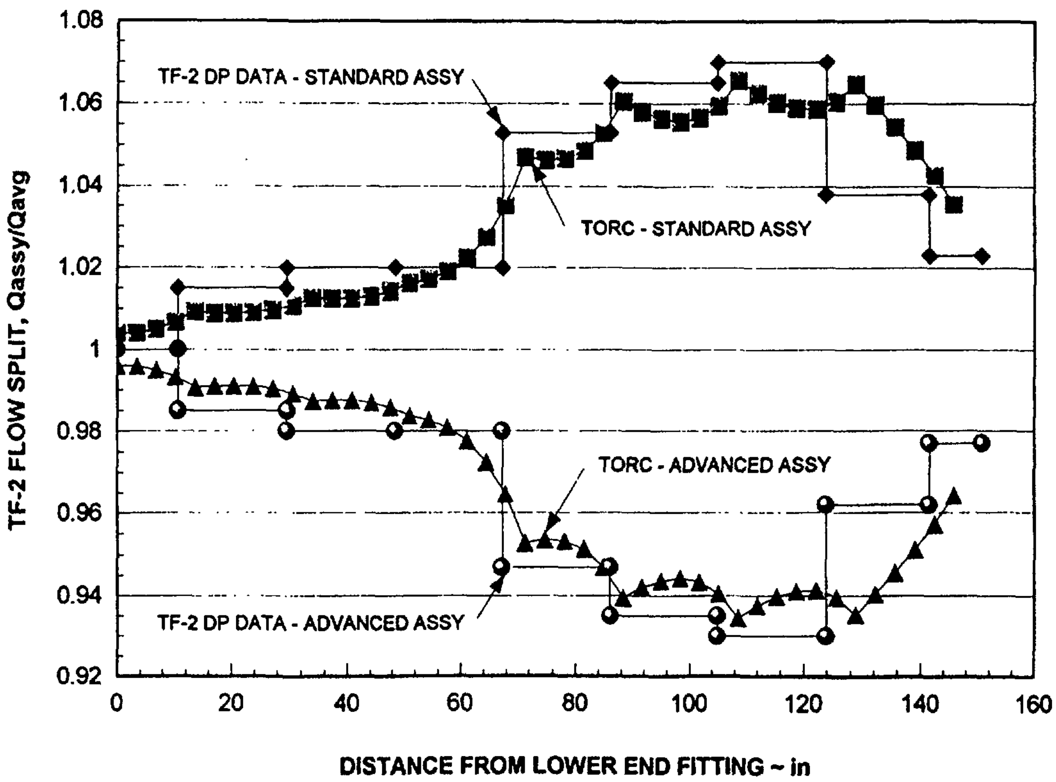

FIG. 4. Comparison of measured flow predicted assembly.

The CFDS-FLOW3D code (now called CFX) was used to predict velocity profiles in a subchannel of a grid span. Figure 5 provides a comparison of the measured and predicted lateral velocities of the swirl in the subchannel 12.7mm downstream of the Side-Supported Mixing Vane grid. As a result of these screening test programs and the CFD analysis, the Side-Supported Mixing Vane grid was selected for the final qualification. CHF testing

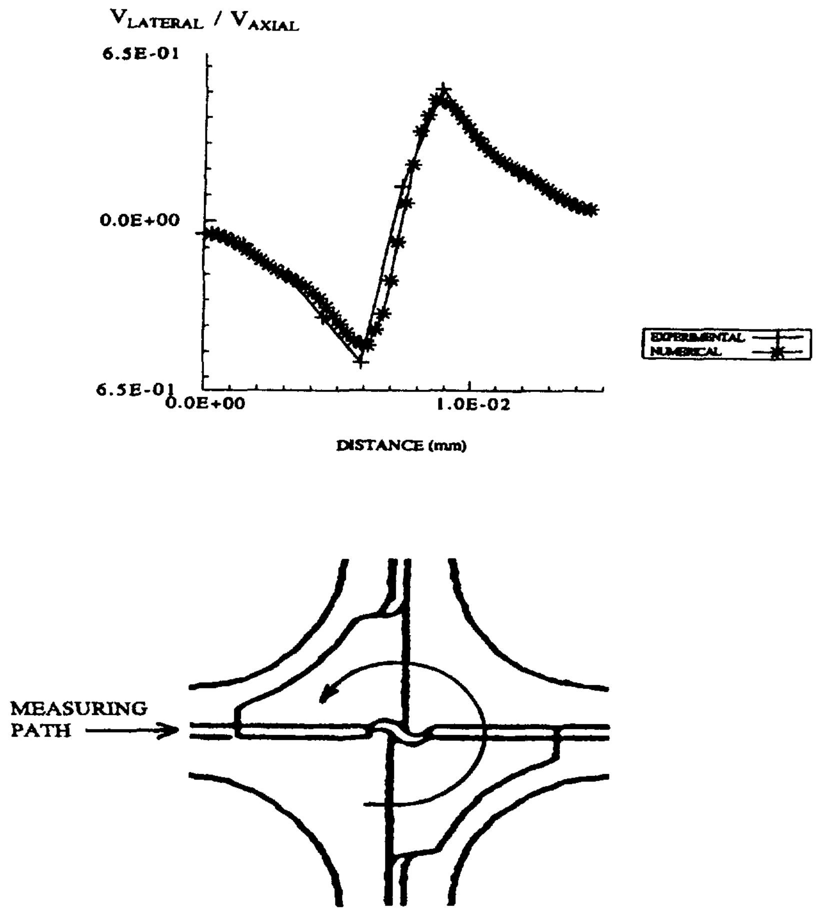

FIG. 5. Subchannel measured and predicted lateral velocity due to swirl from side - supported mixing vanes.

DNB Qualification Testing and Analysis

Critical bundle power at DNB was measured at the Columbia University Heat Transfer Research and compared to the CE-1 CHF correlation [2,3], the correlation for ABB CE's present grid, which has no mixing vanes. The ratio of measured critical heat flux for the advanced grid design to the CE-1 predicted critical heat flux is presented in Figure 6. These test results show an average 44% improvement in DNBR margin relative to a grid with no mixing vanes. Three more CHF tests are planned and then a new CHF correlation will be developed from the data of these tests.

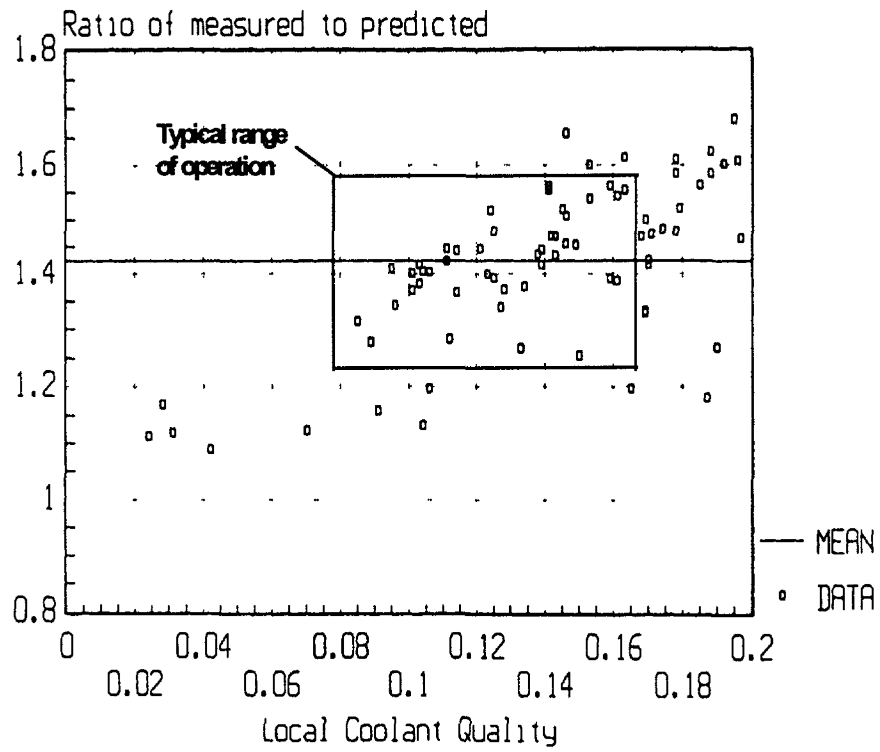

FIG. 6. Improved DNBR Margin for Side-Supported Mixing Vanes

Conclusions

Qualification testing and analyses have demonstrated the superior performance of the Turbo™ fuel design. Lead fuel assemblies containing the advanced features are now undergoing irradiation.

The information set forth herein is general in nature, is not intended to be a specific representation concerning products or services, and does not constitute any warranty or guarantee, express or implied.

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IAEA-SM-353/42P

REGULATORY APPROACH FOR COMPUTER TECHNOLOGY APPLICATION TO SAFETY SYSTEMS IN KOREAN NUCLEAR POWER PLANTS

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1. INTRODUCTION

With the progress of computer technology, the design of nuclear power plants (NPP) safety systems has rapidly changed in Korea to incorporate the new technology. The main objective of this change from analog to computer based is to take advantage of modern computer technology to improve operability and maintainability of the plants. However, from the viewpoint of nuclear plant safety, compared to the conventional analog system, the computer based safety system may be accompanied by new safety concerns such as potential software common-mode failure and/or unexpected failure modes.

The development to date of regulatory positions of the Korean Government on the design of computer based safety system has made substantial progress. However, a lot of controversies have developed between the nuclear utility and regulatory authority over implementation. This paper briefly describes some examples and associated Korean regulatory approaches.

2. STATUS OF COMPUTER BASED SAFETY SYSTEM APPLICATION

2.1. Digital interposing logic system in Korean standard NPPs

The Korean Standard Nuclear Power Plant (KSNPP) is a two-loop type PWR with an electrical output of 1000 MWe. The construction of KSNPP started in December 1989 with Yonggwang units 3 and 4, and began commercial operation March 1995 and January 1996, respectively. As of September 1998, Korea has 8 of these units, such as Yonggwang 3,4,5 and 6 and Ulchin units 3,4,5 and 6, either in operation or under construction.

The NSSS design for KSNPP is basically the same as the ABB-CE's Standard System 80 design. However, the instrumentation and control design incorporates evolutionary features in some parts of the system configuration, such as, the digital interposing logic system (DILS). It receives plant operating commands from the control modules, the on-off logic actuators and the other control systems to provide output signals to field devices, control module indicating lamps, annunciators, and plant computer. The DILS is an integrated microprocessor-based system mainly for control system actuation during normal plant operations that interacts with safety system actuation in case of plant accident conditions.

2.2. Programmable digital comparator in Wolsong Units 2, 3 and 4

Wolsong nuclear unit 2, 3 and 4 are CANDU-PHWR plants supplied by Atomic Energy of Canada Limited (AECL). Each unit has a net output of 700 MWe. The construction of Wolsong unit 2 started in 1991 and commercial operation of units 2 and 3 started in July 1997 and June 1998, respectively.

In the system configuration of CANDU-PHWR plant, there are two independent reactor shutdown systems namely, Safety Shutdown System 1 (SDS 1) and Safety Shutdown System 2 (SDS 2) which has six programmable digital comparators (PDCs). PDC is a microprocessor-based system, programmed to generate the conditioning logic output, signal rationality checks, and variable reactor trip signals for safe operation of the plant.

Functionally, PDCs calculate trip conditions or operating set points for field component actuation. The digital outputs of PDCs drive relays in the channel trip logic used for other trip parameters and internally generated variable set points are displayed on the control room panels.

2.3. Digital data processing and plant protection system in Kori Unit 1

Kori unit 1 is a two-loop type PWR with an output of 587 MWe. It is Korea's first commercial nuclear plant, supplied by Westinghouse, in a turnkey-based contract. Construction and commercial operation of Kori unit 1 started May 1972 and April 1978, respectively.

In system configuration, most of the instrumentation and control systems are composed of analog electronic devices and mechanical relay logic circuitry. Among them, the data processing and plant protection cabinet (Foxboro H-line cabinet) consists of four redundant channels to generate plant protection signals with a 2 out of 4 voting system.

To improve operability and maintainability of Kori 1 unit, the Foxboro H-Line cabinet was replaced with microprocessor-based systems (Foxboro Spec 200/Spec Micro 200) in June - September 1998. The scope of this refurbishment included replacement of electronic modules and data communications related to the NSSS protection system (NPS) and the NSSS control system (NCS). This refurbishment was the first application of computer-based safety system to operating NPPs in Korea.

3. REGULATORY APPROACH FOR SAFETY REVIEW ON COMPUTER BASED SYSTEM DESIGN

The Korea Institute of Nuclear Safety (KINS), a regulatory expert organization empowered by the Government, is responsible for performing the safety evaluation of nuclear power plants and other nuclear facilities in Korea. For the safety review on computer-based safety system, KINS has established the regulatory position that the design requirements of safety systems should be confirmed with the regulatory guidelines of the original vendor country. Especially in the case of operating system upgrades, the licensee should justify the design modification so as to maintain or improve system reliability and safety.

In this context, major safety review concerns of KINS are focused on the following:

- Evaluation of the rationale of design documentation and the design process.
- Verification of the manufacturer's quality assurance program and related documentation.
- Analysis of the system configuration to assure the defense against common mode failure (CMF).
- Audit of the design process and qualification test process when needed.
- Evaluation of the software verification and validation (V&V) efforts and related documentation.
- Confirmation of the software configuration management (CM) program and system maintenance program, etc.

IAEA-SM-353/43P

DESIGN CONCEPT OF A LIGHT WATER COOLED PHWR

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The design concept of a pressure-tube type, light water cooled, heavy water moderated reactor was proposed as an evolutionary Light Water Reactor (LWR) based on three design prototypes: CANDU, PWR and PLPTR [1].

DESIGN CONCEPTS

A reactor design concept, shown in Figure 1, is based on proven technology of CANDU with CANFLEX or DUPIC fuels. The geometrical shape and size of pressure tubes and fuel bundles are to be the same as those of CANDU. Therefore, the same NSSS and BOP of CANDU plants could be used. Light water instead of heavy water as a coolant would have no impact on thermal-hydraulic safety limitations, and the same geometry of fuel bundles would be compatible with a CANDU fuel reloading machine. This concept has some favorable features compared with PWR and CANDU. First, the use of pressure tubes instead of a pressure vessel for the RCS gives a more intact system boundary and the possibility of on-power refueling for LWR. The separation of pressure tubes in a calandria tank provides a large space between them for the installation of reactivity control and core monitoring devices. This feature extends the capability of automatic control with a versatile control system and on-line maintenance of monitoring systems.

The use of a light water coolant requires enriched uranium fuels, with a fuel cycle cost burden. However, we can eliminate the tritium production problem and also reduce the amount of spent fuels produced in CANDU. Water-to-water interface in the steam generator is an another benefit. Daily fuel loading makes the core excess reactivity low enough to eliminate the soluble boron control requirement and this makes water chemistry more favorable than that of the PWR.

To compensate for the reactivity loss from light water coolant, heavy water moderator should enforce neutron moderation. In CANDU, large volume of heavy water in the calandria tank causes a design limitation in the scale-up. As shown in Figure 1, heavy water moderator is contained inside the secondary calandria tube. A gas filled insulating layer, inner tube, moderator layer, and outer tube surround the pressure tube. Outer spaces of the calandria tubes are left vacant in the dry calandria concept. This double tube is effective for structural enforcement against tube sagging.

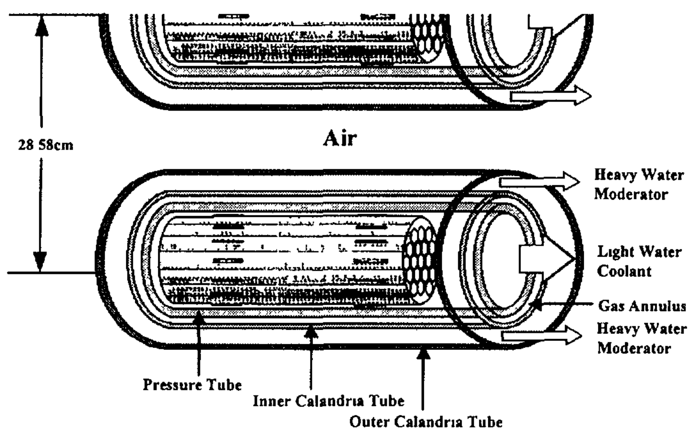

Fig 1 Design concept of pressure tube reactor

Passive safety features are not included in this reactor design concept, however, emergency cooling and long term cooling can be achieved by passively flooding the system using the dry calandria concept. Emergency light water can be drained into a vacant calandria tank by gravity, either by active or passive valve opening. In a pressure tube break, pressurization of the calandria tank can be avoided by pre-pressurizing moderator tubes around the coolant pressure tubes and passive flooding provides large amounts of water as a thermal sink and additional shutdown system.

A major benefit of CANDU is the feasibility of automatic power control. Because enriched uranium fuels are used in this study, excess reactivity of the unit fuel bundle is much higher than one of natural uranium fuels. When excess reactivity is controlled to equal that of CANDU fuels, the control system can be used without change. A burnable poison can control excess reactivity of the fuel bundles.

FEASIBILITY OF NUCLEAR DESIGN

After a parametric study done by a unit module calculation with a code system HELIOS [2], an optimal reactor module was found, as shown in Figure 2. With a 2.0-cm thick layer of heavy water moderator, all feedback coefficients were found to be negative. Light water coolant temperature coefficients were -25 to -30 pcm/K, heavy water moderator temperature coefficients were -7 to -9 pcm/K, and void coefficients were about -100 pcm/K in the wide range of fuel enrichment levels and fuel burn-up steps. Figure 3 shows the excess reactivity letdown curves for the designed reactor module and CANDU. The uppermost line in this figure is much higher than the lower line for CANDU; this difference of excess reactivity should be controlled with a burnable poison. The conventional CANDU fuel bundle has 37 fuel pins with one located in the center and 6, 12, 18 pins at three rings. Without correction, power generation densities throughout the fuel pin rings are not small, about 1.8 times. ZrB_2 coating as IFBA concept was used for a limited number of fuels pins at the outer ring locations. A 0.0254-mm coating to three fuel pins makes the excess reactivity curve be fit to one of CANDU, as shown in Figure 3. From this figure, we can conclude that discharge the burn-up length of 2.8 w/o SEU fuel bundles is about 2.4 times that of conventional CANDU fuel bundles (about 7500 MWD/MTU).

Figure 4 shows a shutdown capability of water flooding into a void region of the dry calandria tank. In this figure, the negative reactivity insertion due to light water injection was more than 50% throughout the fuel burn-up.

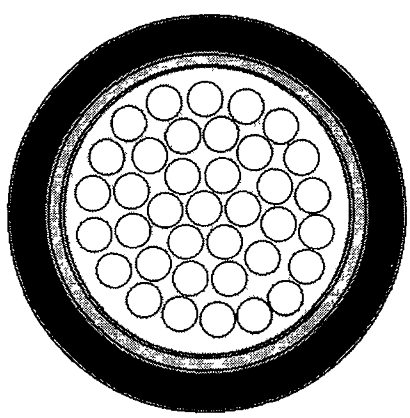

Fig. 2. Cross sectional view of unit reactor module.

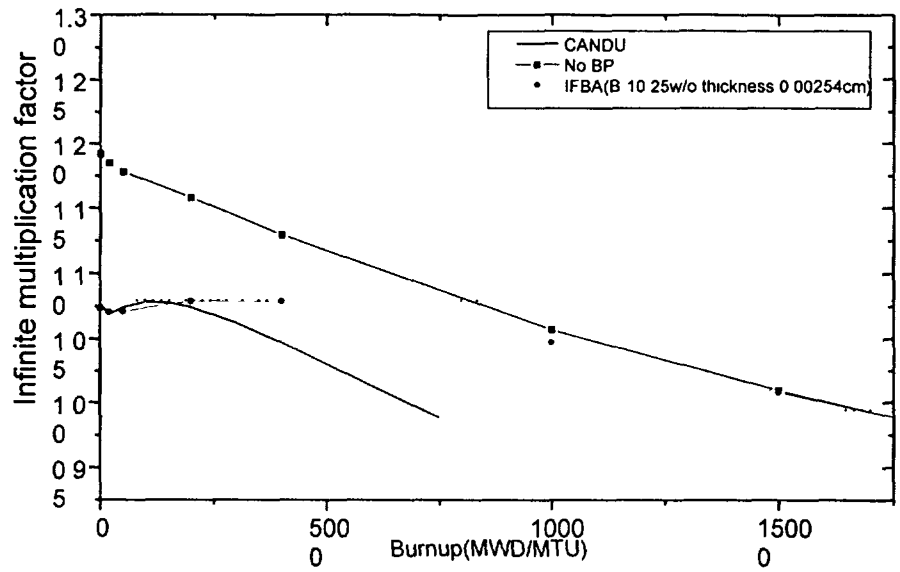

FIG 3 Excess reactivity change of rx module & CANDU

SUMMARY AND EXPECTED ECONOMICS BENEFITS

Optimized design parameters for this reactor concept are 2.8 w/o enriched uranium fuel, light water coolant, and a 2.0-cm thick heavy water moderator layer within double calandria tubes. This design has all negative feedback and additional shutdown safety by water injection into dry calandria tanks As a result of excess reactivity control by IFBA burnable poison, CANDU control devices can be applied without design change, extending the fuel discharge interval 2.4 times to that of CANDU. The nuclear characteristics of this reactor module concept are concluded to be feasible and would have no major impact on the CANDU thermal-hydraulic and MMIS design.

The increased fuel cycle cost for enrichment compared to CANDU may be compensated somewhat by the reduced amount of spent fuel production and reduced refueling work load expected from an extended discharge interval. Reduction of heavy water in a calandria tank would be of great benefit both in capital and O&M costs. Even though the double calandria tube design is more complex, it has greater structural integrity and results in a higher potential for plant size scale-up. Compared to the PWR economics, on-power refueling capability in LWR would be a major incentive.

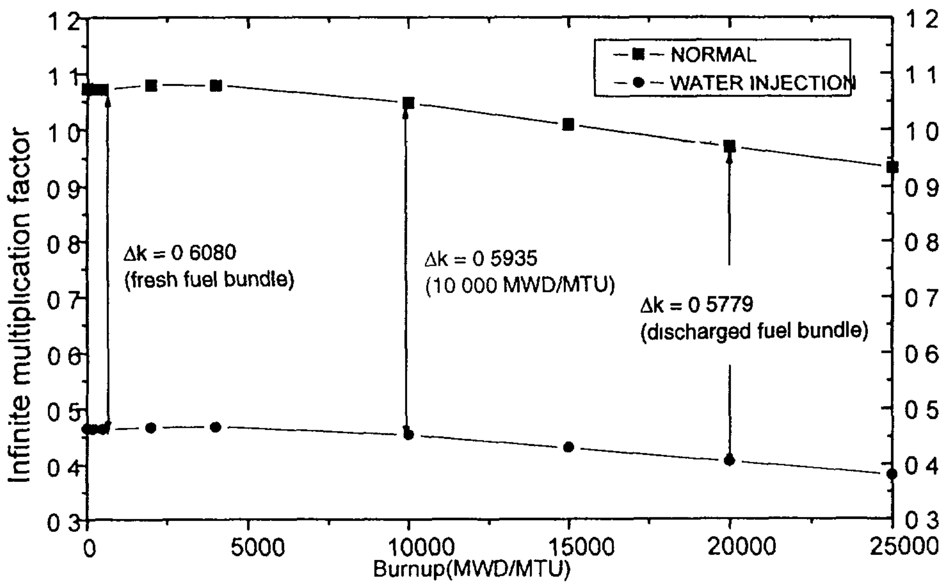

FIG 4 Change of reactivity by water injection to dry Calandria

ACKNOWLEDGEMENTS

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IAEA-SM-353/44P

EXPERIMENTAL INVESTIGATION OF MIXING PHENOMENA INSIDE A CONCRETE CONTAINMENT COOLED BY AN INNOVATIVE PASSIVE SYSTEM

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The paper describes the Large Scale Test Containment Facility located at the Westinghouse Science & Technology Center and modified to simulate in a approximate 1/10th scale the main features of the ENEL concrete containment; the heat is removed through internal heat exchangers (HX) located in the dome region and external HXs placed outside the double barrier concrete containment, connected by an intermediate fluid loop. No active component like pumps nor human intervention is required for the operation of the system.

The facility instrumentation, the test program and the experimental results are described along with the first results obtained in the application of the FUMO code in the analysis of the presented experimental tests. The measured data include the temperature distribution inside the containment under different test conditions, examples of helium concentrations at four internal locations, and laser doppler anemometer measures to determine the atmosphere mixing.

ANALYSIS OF THE EXPERIMENTAL RESULTS

The steam-air concentration measured at different locations in the experimental apparatus are correlated in a physically-based map proposed for the analysis of the atmosphere stratification. In this map the normalised steam flowrate is reported as a function of the concentration gradient index (defined as the normalised difference between the molar air concentration measured below the operating deck and in the upper dome) (Fig. 1). To get more insight about the time dependence of the flow evolution inside the facility, a stratification ratio has been defined taking the ratio between the concentration gradient index and the normalised steam flowrate. The M02-Run36 Test simulates medium power (60 kW) and medium pressure (297 kPa) conditions inside the LSCTF. Figure 2 shows the atmosphere pressure trend during the test: the time window during which helium is injected is identified and the related helium distribution at four locations inside the apparatus is also reported. The dashed curve shows the prediction of pressure trend obtained with the FUMO code: it is satisfactory except a slight overestimation during the first steady-state period. Helium equalises quite rapidly in the large space over the operating deck, while the concentration below the deck increase more slowly, reaching the equilibrium value after four hours. The time trend of the stratification ratio between the dome and the space below the operating deck (curve bd-dome in Fig. 3) shows high stratification values only up to 11000 s., after which a slow but continuous homogenisation of the atmosphere is clearly detected. The corresponding stratification map related to the "bd-dome" curve is presented in the Fig. 4 (the marked points of the map show the phase after the injection of helium). The preconditioning of the facility (lasting from the beginning of the test to until the time C) is achieved by injecting large steam flowrates, which cause the sweeping of a large fraction of the air from the upper region to the space below the operating deck. This stratification persists for about 1000 s (phase C-D-E) during which the steaming is strongly reduced to reach the desired steady-state conditions, followed by a slow mixing (phase E-F-G) of the atmosphere between the upper and the lower regions of the containment. This process is clearly associated with a different flow paths inside the facility, characterised by convective motions between the two regions of the apparatus (air coming up, steam and helium coming down), which can explain the slow but continuous homogenisation of helium concentration below the operating deck.

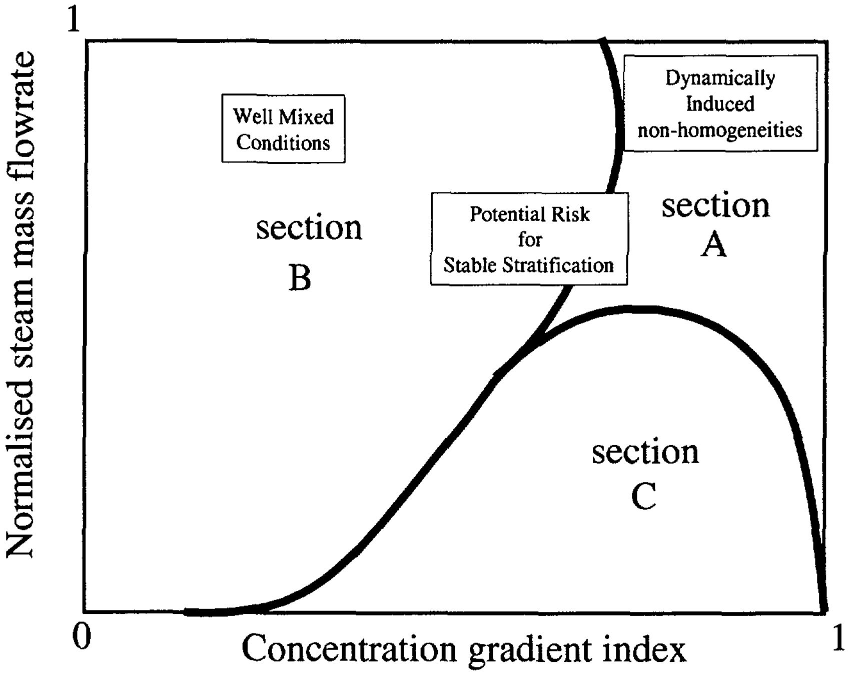

FIG. 1. Stratification map example.

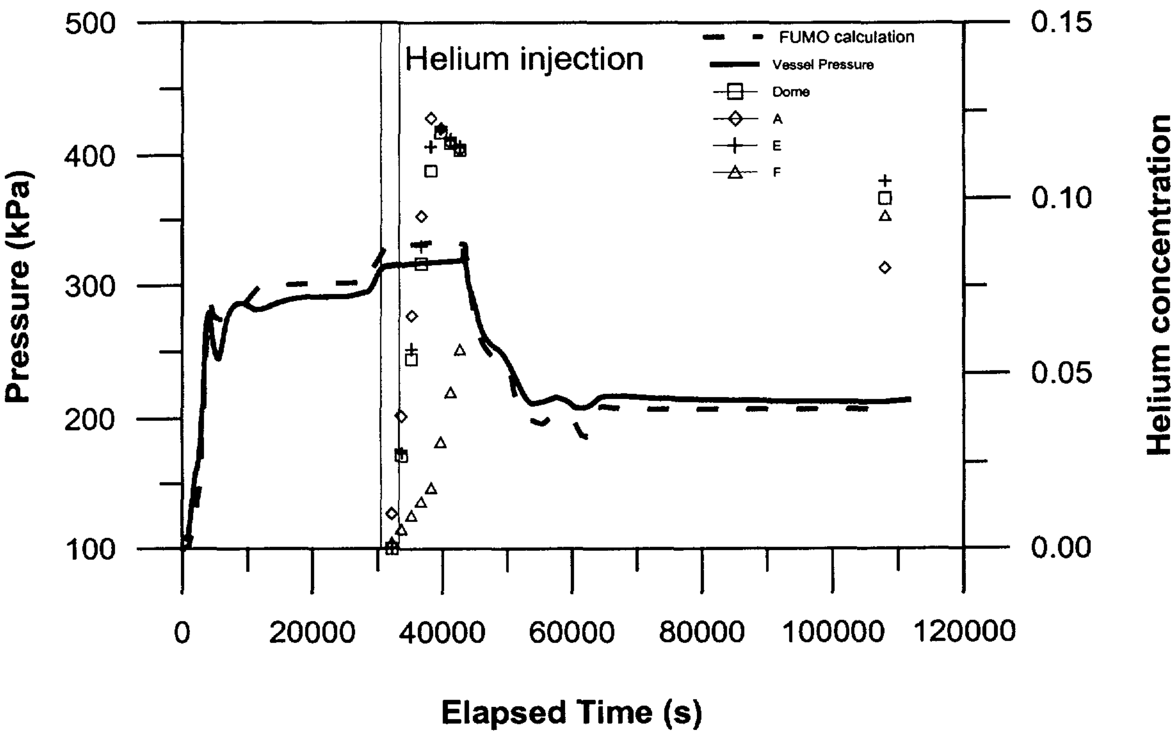

FIG. 2. Test M02-Run 36 : Vessel pressure and helium concentrations.

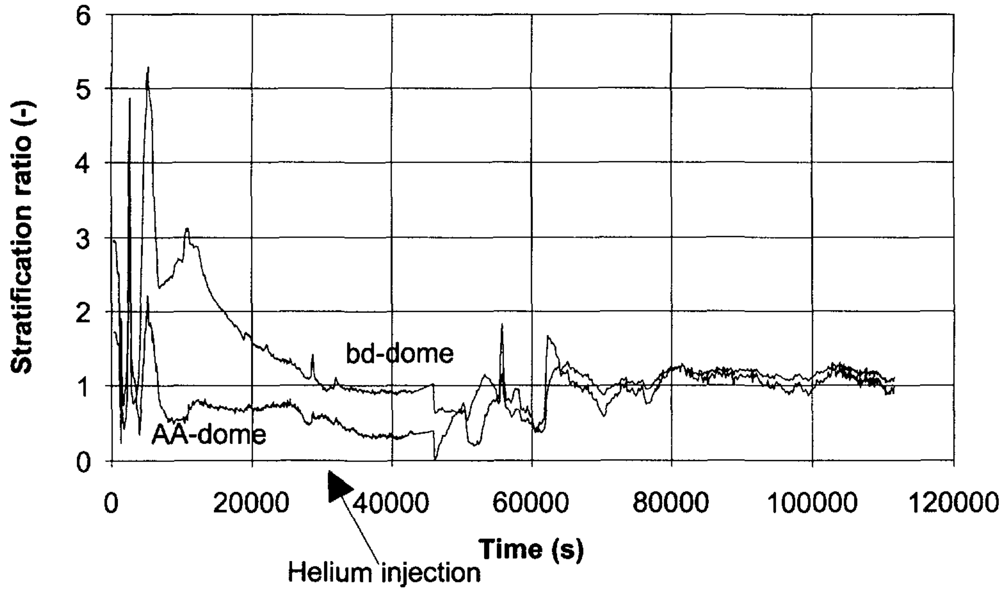

FIG. 3. Test M02-Run36 : Stratification ratio vs time.

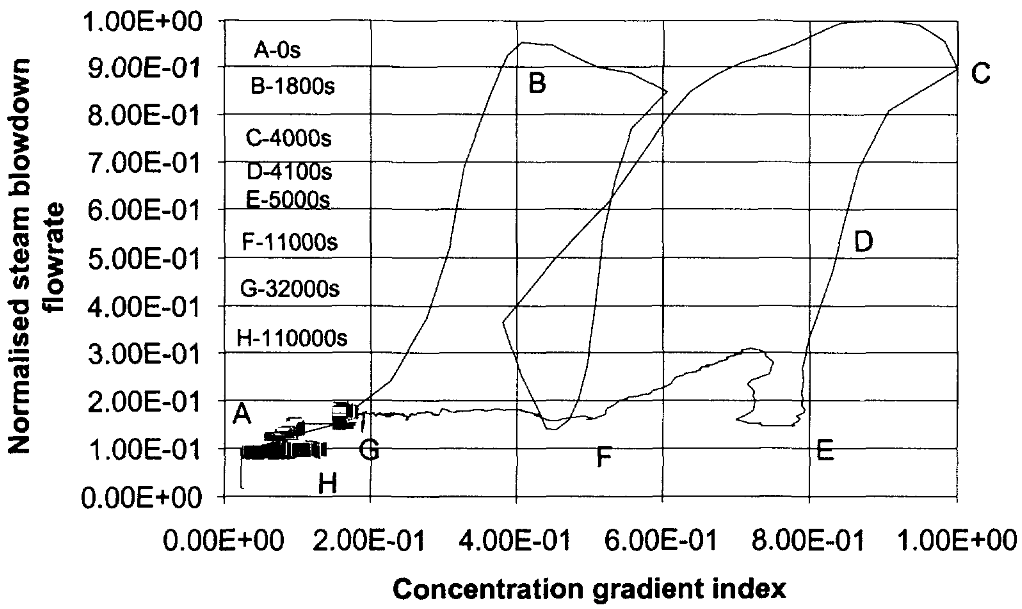

FIG. 4. Test M02-Run36 : Stratification map.

CONCLUSIONS

The ENEL test campaign on the Large Scale Containment Test Facility installed at the Westinghouse Science & Technology Center allowed to test in an integral way the behaviour of an innovative system in which the heat is removed through internal heat exchangers (HX) located in the dome region and external HXs placed outside the concrete containment, connected by an intermediate fluid loop, and not requiring any active component for the operation.

The experimental results of the test campaign, particularly focused on the characterisation of the atmosphere mixing inside the containment during a wide range of test conditions, confirmed a good performance of the proposed solution both in term of the cooling capability and in term of avoiding dangerous stratification inside the facility. In all the investigated test conditions the helium undergo a near homogeneous distribution in the facility, as represented by characteristic states belonging to a well mixed region in the stratification map. This very good mixing experienced by the helium, indicates that the localised PCCS promotes efficient convective motions inside the containment atmosphere, giving positive indication for related safety concerns. The parallel analytical work performed as a support to the experimental activity confirmed the prediction capabilities of qualified containment codes, providing a powerful tool for the investigation and comprehension of the considered complex phenomenology.

IAEA-SM-353/45P

PASSIVE CONTAINMENT COOLING SYSTEM (PCCS)

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Containment system plays a fundamental role in the management of accidental conditions, including Severe Accidents.

One of the objectives of the reactors of the new generation is to strongly limit the environmental impact of the plant, even in the unlikely event of a severe accident, so that immediate protective countermeasures outside of the site boundary are not needed.

The European Utilities Requirements (EUR) ask for a containment which is able to mitigate external hazard effects, including airplane crash and pressure wave, and internal effects of hypothetical severe accidents, maintaining a stringent leaktight for the whole accident duration. In addition, EUR asks for a double containment as the preferred solution.

As a consequence, one of the preferred containment configurations for European future plants, but also in other country as Korea, is the double concrete containment.

This solution makes the decay heat removal from the containment a challenging issue. Different solutions for decay heat removal from the concrete containment have been studied by ENEL together with ANSALDO and ENEL R&D Division before selecting the one which is able to meet the following conditions:

- Heat removal
  - primary containment structural integrity
  - primary containment design leaktightness
  - pressure reduction to 50% of the containment design pressure within 24 hours
  - "Severe Accident Safe State" (SASS)
  - long-term concrete integrity
- Hydrogen control function
- Primary and Secondary containment isolation function
- Source Term reduction

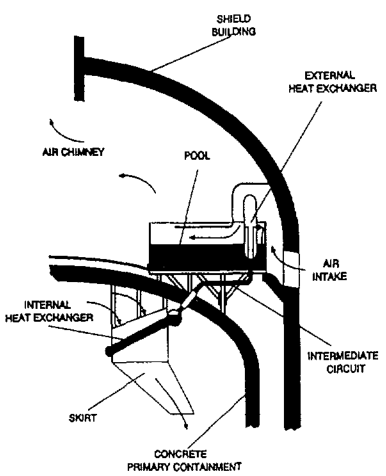

FIG. 1. PCCS schematic.

The innovative containment decay heat removal system for double concrete containment consists of a number of two-phase (water-steam) closed loops in which the non-condensables are removed. Each loop is provided with internal and external heat exchanger (immersed in a water pool) and connecting piping (fig.1).

The number of loops depends on the plant size and on the containment design pressure.

Non condensables are practically absent in order to allow the system operation also at low containment atmosphere temperature, to avoid freezing for the external water pool during severe weather conditions and to allow a continuos detection of the circuit integrity and a higher heat removal efficiency.

The internal heat exchanger, located in the containment dome, consists of finned tubes connected to upper and lower headers. The tubes are aligned in a small number of rows staggered and inclined. The containment atmosphere circulation through the tube bundle is driven by the natural draught of a skirt (reverse chimney).

The external heat exchanger consists of "U" shaped finned tubes with a vertical common header for both steam inlet and condensate drain outlet. The external heat exchanger is immersed in a pool; cooling air driven by an external chimney flows on the pool surface after the water level decrease.

The heat transfer mechanisms are:

Inside containment:
Condensation of the steam-air mixture inside the containment atmosphere on the external surface of the internal heat exchanger, boiling of the cooling fluid in the intermediate loop (internal side of the internal heat exchanger tubes).

Outside containment:
Heat transfer of the cooling steam-water mixture and condensation in the external heat exchanger. Heat is transferred to the pool water up to the boiling temperature. Then, decrease of the water pool level due to the boil off starts a natural draught air circulation; in this phase, heat removal occurs by the combination of air cooling on the dry portion of the heat exchanger and water pool evaporation. In this way the pool water temperature and the containment pressure start decreasing. The pool water capacity has been sized for more than 3 days of operation with neither operator action nor active system being required. Finally, when the pool is empty, the external heat exchanger works as an air-cooled heat exchanger if water is not restored; in fact, the external heat exchanger has been sized so that the heat removed by air matches the decay heat at 72 hours keeping the containment pressure well below the containment design pressure.

The air flow is prevented by the water, which acts as a water seal, during normal operation. This water seal will avoid the water freezing and removal of excessive heat from the containment during normal operation with severe cold weather conditions.

Furthermore, an option of this innovative system provided with spraying system before 72 hours has been also tested. Water from the pool can be used for cooling the dry portion of the external HX in order to increase the heat transfer and, then, to reduce the containment pressure up to a value close to the normal range in about one week. This spraying system is more economical and, at the same time, the electrical power required is very low so that a sufficient independence of the containment heat removal system from the preventing systems can be maintained (e.g. by means of dedicated batteries).

An extensive research programme has been set-up and funded over the last years by ENEL in order to support the design effort. Part of this test campaign has been developed, from 1996 until 1998, in the INCON (INnovative CONtainment Cooling for Double Concrete Containment) Programme supported by the European Commission in the frame of the Fourth Framework Programme on Nuclear Fission Safety; the Programme has been carried out by a Consortium of 6 Organisations (ENEL, ANSALDO, CIRTEN, Empresarios Agrupados, EdF-SEPTEN and PSI) under the co-ordination of ENEL.

The test programme included separate effect tests and integrated test of the whole system to better investigate the interactions among different phenomena.

In particular, the experimental test campaign included:

- thermal-hydraulic tests on single finned tubes and on bundle of the internal HX
- thermal-hydraulic tests on the intermediate loop

- thermal-hydraulic tests on the external HX
- thermal-hydraulic tests on small-scale integral system
- natural circulation test
- corrosion tests
- analytical activity

At the end of this process an innovative containment heat removal system for decay heat removal from a concrete containment, which is applicable to all the containment types and compatible with the economical goal, plant layout and operational and maintenance requirements, has been selected for design development.

CLOSING SESSION

Chairperson

P.E. JUHN
IAEA

STATEMENT BY SESSION CHAIRPERSON

P.E. Juhn
International Atomic Energy Agency,
Vienna

Distinguished participants, ladies and gentlemen: My name is P. E. Juhn and I am the Director of the Division of Nuclear Power of the IAEA. It is my honour to chair this Closing Session on behalf of IAEA.

On the first day of this International Symposium on Evolutionary Water Cooled Reactors, we have heard on the first day very encouraging and important messages in the Opening Session from three distinguished speakers:

Kang, Chang-Hee, Ministry of Science and Technology, Republic of Korea
M. ElBaradei, Director General, IAEA
Jang, Young Sik, President and CEO of KEPCO, Republic of Korea.

As Dr. ElBaradei, Director General of the IAEA, pointed out, global awareness of the need for sustainable energy strategies provides the context for recognition of the role of nuclear power as an environmentally friendly source of energy. However, even though nuclear power provides about 17% of world electricity supply, which accounted for the avoidance of about 8% of global carbon emissions, the future for nuclear power presents a very mixed picture. Dr. ElBaradei noted that "there are compelling reasons why nuclear power should remain an important source of global electricity supply, but in many countries there are also substantial hurdles in terms of public acceptability and economic competitiveness. In the absence of viable alternatives, the world would not be well served if an important energy option were rejected. The challenge is to ensure that the nuclear power option is given a full and fair hearing. Meeting this challenge requires a two track approach. One track is to continue to work on improving the operational safety and efficiency of existing nuclear power plants. The second track is to foster research and technological development with the aim of constantly improving every component of current nuclear fuel cycle technology while also developing advanced evolutionary and innovative reactor designs".

During the past 3 days we have heard presentations on the factors important for decision-making with regard to building new evolutionary water cooled reactors, such as:

- economic competitiveness, technological advances and good performance, which are receiving strong attention from design organizations;
- financing challenges and solutions;
- social-political conditions;
- institutional preparedness;
- safety requirements and international co-operation in nuclear safety.

We have learned about recent technical advances with respect to:

- decay heat removal;
- containment arrangements;
- achieving balances between active and passive safety functions;
- new I&C equipment and systems; and
- prevention and mitigation of severe accidents.

These factors and technical advances are important considerations of governments, regulators, and utilities.

We have also been briefed by design organizations on key developments and the status of quite a number of evolutionary water-cooled reactor designs; the technology and technical solutions seem to be well established. Here, I may add that we have also had an opportunity to have a more detailed look into these designs in the exposition area, and through the technical information provided by the design organizations in their design description papers.

Earlier today, this morning and this afternoon, we have had a review of "keys to economic viability", ranging from the merits of standardized designs and proper management policies, construction achievements, feedback of operating experience into new designs and utility requirements, and the optimisation of the fuel cycle. In addition, we have had an opportunity to familiarise ourselves with some special design features and development activities during the poster sessions.

In this Closing Session, have invited our distinguished panelists to address key issues including the economic challenges, safety objectives, the role of nuclear power in sustainable development, infrastructure requirements, financing approaches, and the role of international organizations.

At this let me introduce our distinguished panelists, in alphabetical order by country:

D. Torgerson	Vice-President, Research & Product Development, Atomic Energy of Canada Limited. Also he is serving IAEA as a chairman of the new International Working Group on Heavy Water Reactor Technology , since its formation in 1997	Canada
W. Zeng	Senior Vice President, Guangdong Nuclear Power Company	China
A. Birkhofer	Managing Director, Gesellschaft fhr Anlagen - und Reaktorsicherheit, also is the chairman of IAEA's INSAG. Mr. Birkhofer also made an excellent speech at a special occasion this morning, not related to the symposium, to the Korean Chapter of American Nuclear Society and Korean Nuclear Society.	Germany
A. K. Anand, representing, A.Kakodkar	Mr. Anand is Director of the Reactor Projects Group, Bhabha Atomic Research Center, representing Mr. Kakodkar, Director of Bhabha Atomic Research Centre who was unable to attend	India
Y. Oka	Professor, Nuclear Engineering Research Laboratory, University of Tokyo	Japan

C.S. Kang	Professor, Department of Nuclear Engineering, Seoul National University	Republic of Korea
Y. Dragunov	Chief Engineer of the Design Bureau, OKB Gidropress (Mr. Dragunov was unable to attend)	Russian Federation
W.D. Magwood IV	Director, Office of Nuclear Energy, Science and Technology, US Department of Energy	USA

Next we will have two presentations to brief the panel on the key points from the plenary sessions, and from the poster sessions.

BRIEFING TO THE PANEL ON KEY POINTS OF PLENARY SESSIONS

J. Kupitz
International Atomic Energy Agency,
Vienna

Distinguished chairperson, distinguished panelists, ladies and gentlemen: I have the honour to present to you some background information about the key issues and results achieved during the plenary part of this symposium.

We are all aware that the increase in population, industrialisation and the need to increase the standard of living — in particular in developing countries — requires more energy, in particular electricity. Today about two billion people are living without any access to public electricity grids; so we have a mandate here. Energy and electricity can be provided by various sources, but nuclear energy and to a certain extent hydro-electric power are the only sources which can provide these energies in a safe, reliable, economic condition and under environmentally benign conditions. The potential for hydro-electric power in several countries has already been fully implemented.

Water cooled reactors — the subject of this symposium — represent by far the majority of all operating reactors and reactors that are currently under design. To compete with other energy options, nuclear energy has to reduce the investment costs, it has been stated, up to 50%. Also current designs should be supplemented by other designs in the small and medium sized category for countries with smaller grids, for remotely located areas, or for specific applications such as district heating and in particular seawater desalination for the production of potable water.

It was reported that nuclear power is mostly expanding here in Asia. Thirty percent of the electricity generation in Japan is already produced by nuclear energy and here in the Republic of Korea, in 1997 more than 35% of the electricity was generated by nuclear power. Currently, this value has risen to over 40%. Despite the well known economic crisis which is currently prevailing, the outlook for economic growth and the additional demands for energy remain positive in this area. For example, it was reported that China plans to construct 18 GW(e) of additional capacity by the year 2010.

Nuclear energy is considered by the nuclear community as a sustainable energy source. This conclusion has to be substantiated and all efforts are to be made that this conclusion is accepted by the non-nuclear community. The IAEA is highly interested to support such a global acceptance of this conclusion.

World wide about 700 million US dollars are spent per year for technology development and design for water cooled reactors. This rather big amount of money is being spent at various institutions including: the design organizations themselves, research institutes, safety authorities and universities. The funding comes from both the governmental and also the private sector. Most of these 700 million US dollars per year are spent on evolutionary water cooled reactors which are being developed with the objectives of reduced costs, high availability while meeting at the same time increasingly stringent safety objectives.

For nuclear energy, as we are all aware, strong competition exists, in particular, from combined cycle gas fired power plants. To meet this competitive challenge, several measures should be taken, including: cost reduction, short construction periods, a stable regulatory environment, simplified and standardised designs with low capital costs are preferred. Financing of nuclear power is an issue that has or will be solved in industrialised countries In developing countries the challenge of financing may be met by a combination of loans from the supplier countries and domestic loans.

Socio-political factors have a strong influence on public acceptance. The nuclear community needs to take all measures to address concerns of the public and to highlight the benefit of nuclear energy.

With regard to safety principles for nuclear power, they are well known and being harmonised. There is already an increased international co-operation in nuclear safety, for example in the framework of the convention of nuclear safety and the activities of WANO to share imported information among all operators of nuclear power plants in the world. A continued dialogue between utilities and vendors has resulted in utility requirements documents which specify utility requirements for future reactors. In this context I also would like to point out that the IAEA has prepared a guide, in particular for developing countries, to establish their own design requirements. This is being published now.

Good advances in technology development for safety have been reported. New systems have been developed and tested to remove the decay heat from the reactor core. Each design organization carefully evaluates active and passive systems and selects the best and most economic system to meet the required safety function. Most evolutionary designs incorporate features to reduce the probability of severe accidents and to mitigate their consequences. All evolutionary designs take advantage of modern instrumentation and control technology.

There was much good news reported during this symposium. Many evolutionary reactors were reported to be moving through the detailed design phase. Methods are being identified to reduce the capital cost. As examples, the EPR of Nuclear Power International, the System 80+ of ABB Combustion Engineering, and also the Korean Next Generation Reactor, have identified increasing the plant electrical output as one option for reducing the capital cost per KW(e). output. Other designs, for example the Siemens SWR 1000 boiling water reactor, incorporate passive safety systems to achieve simplification and a resultant reduction in capital cost.

Furthermore, some design organizations have reported completion of successful licensing reviews and licensing certifications. As examples, in the USA, the AP 600 has received Final Design Approval, which is the step before Design Certification, from the US Nuclear Regulatory Commission (US NRC) in September 1998. The System 80+ of ABB Combustion Engineering, and the Advanced Boiling Water Reactor of General Electric have received Design Certification from the US NRC. The evolutionary CANDU-9 design of Atomic Energy of Canada, Ltd., has received a positive licensing review by the Canada's Atomic Energy Control Board. And here in the Republic of Korea the Korean Next Generation Reactor is to be submitted in 1999 for design certification. Also in other countries there are ongoing efforts in evolutionary reactor designs. In China the experience from the Qinshan 300 MW(e) reactor is being used in the design an evolutionary plant with passive safety systems — the AC 600. Our colleagues from India are designing a heavy water reactor plant which will incorporate the thorium fuel cycle, and also utilizes new passive safety systems for the core cooling and the removal of decay heat.

Positive experience from standardisation was reported by Electricité de France. The manufacturing of series of components and the learning effects of repetitive construction procedures yield significant cost reductions. Key lessons in plant management have also been reported by Ontario Hydro. This experience includes the importance of having the right people with the right qualifications in the right place at the right time. The need for all staff to have a questioning attitude and to be committed to a safety culture, good configuration management and a high priority on maintenance was reported as being very essential. We appreciate Ontario Hydro's willingness in sharing this experience and giving us good guidance for other utilities to deal with their nuclear power plants. Good news was also reported regarding the construction time needed for nuclear power plants in particular from TEPCO in Japan. The Kashiwazaki Kariwa advanced boiling water reactor units six and seven were finished in very short periods.

To achieve good performance there is a very close interaction between various design organizations and utilities. This has been reported for the USA, for Europe, for Canada, India, the Republic of Korea and other countries. And lessons learned during constructions and operations are fed back immediately into current and next step designs. It was also reported today that various fuel cycle options are available for the evolutionary water cooled reactors. They can be optimised subject to a wide range of criteria of which economics, sustainability of resources, environmental aspects and proliferation resistance are dominant along with special national objectives in the individual countries. And there is no single strategy optimal for all countries.

BRIEFING TO THE PANEL ON KEY POINTS OF POSTER SESSIONS

K. Foskolos
Paul Scherrer Institute,
Switzerland

Mr. Chairman, distinguished panellists, ladies and gentlemen: I have the honour to summarize the key points from the poster sessions. Along with the invited papers on strategic issues determining the future environment for the deployment of evolutionary water cooled reactors, this symposium called for poster presentations which should give concrete examples of current efforts and prospects of technological developments in nuclear industry and research. Forty onse out of fifty one submitted poster abstracts were selected by a committee according to criteria of scientific quality, novelty of information and relevance to the symposium. These posters have been presented during the last two days and I will in the following summarise briefly their contents. In particular, I will focus on some of them which were highest rated during the selection process by the committee. The presented posters can be divided in thematic categories which do not necessarily correspond to the order of their presentation during the two poster sessions.

One category of posters dealt with general energy and nuclear power generation policies and economic aspects. Six poster papers from Argentina, Belarus, Indonesia, Romania, Switzerland and Ukraine illustrated from different angles strategic issues regarding the future deployment of nuclear power world-wide. While the posters from Belarus, Indonesia and Ukraine focused on the necessity to implement in the respective countries nuclear power plants using evolutionary water cooled reactors, and described the problems encountered and possible means to overcome them, the Romanian poster concentrated on the economic problems related to the completion of the second Cernovoda unit and sought for an advanced financing scheme.

The Argentine poster resumed an older idea from the sixties, namely water cooled reactors with their secondary stem flow overheated by the exhaust gasses of a gas turbine, thus combining nuclear power and modern gas fired plants in a strategic partnership. This option was once abandoned because of technical problems at that time –specifically, low load factors, and materials problems — that do not exist anymore. Recent studies show that for a majority of countries the present level of natural gas prices and the capital costs for new nuclear power plants lie within a range which makes this symbiosis viable and more interesting than a strategy based on either of the two technologies individually.

The poster from Switzerland dealt with ecological and risk related performance of nuclear and other electricity generating systems based on life cycle analyses of different electricity generating systems that reflect the impact of anticipated technical progress in a 20 to 30 years perspective. The studies demonstrate among others the clear ecological advantages of nuclear and hydro power, for instance regarding green house gas emissions, although a reduction by a factor of up to two can be expected for advanced natural gas systems in the future. Regarding the risk from severe accidents, one of the most interesting insights resulting from an extensive data base covering the period 1969-96 is the evident difference between OECD and non OECD countries.

The next category of posters described research and development for evolutionary water cooled reactors. There were nine posters from Canada, Germany, Italy, the Republic of Korea, Switzerland and the USA, that provided examples of current fundamental and industrial research, and the development of specific aspects of evolutionary water cooled reactors. The Canadian posters dealt with modern methods in the construction and design for CANDU reactors. The Italian posters dealt with passive containment cooling systems engineering and with PCCS mixing tests in the containment atmosphere. The posters from the Republic of Korea covered a range of topics, including the development of human factors evaluation techniques for nuclear power plants, experimental analysis of passive hydraulic valve characteristics, and the in-vessel retention against core melting accidents. The USA poster described the present status of test programmes for advanced fuel designs for the ABB Combustion Engineering and the KSNP reactors.

The German poster described a recently implemented network of safety related innovative nuclear reactor technology R&D in European countries. This network named SINTER, is an activity within the 4th Framework Programme of the European Commission which aims at gathering information on current and proposed R&D to streamline efforts and to optimise the distribution of shared research work and corresponding funds among institutions. Both a bottom up approach — that means proposals from research organizations — and top down approach — that means policies and priorities of governments, industry and safety authorities -are used. Data is collected with Internet-based interactive questionnaires.

The Swiss poster dealt with the ALPHA programmes and the corresponding thermal hydraulic facilities at the Paul Scherrer Institut. The ALPHA project was initiated 1991 for the large scale integral experimental and analytical investigation of long term decay heat removal and fission product retention. In the current project phase, ALPHA-II, components of General Electric's ESBWR and Siemens' SWR 1000 were investigated at the two main facilities PANDA and LINX and their expected behaviour could be demonstrated.

A third small category of posters dealt with nearly commercial evolutionary water cooled reactors. This category included five posters from Canada, France and the Republic of Korea that dealt with the current optimisation work of the nuclear industry on nearly commercial reactor concepts. The Canadian poster provided a concise survey of the inherent safety features of the CANDU reactors. The posters from the Republic of Korea addressed the optimal elevation of seismic support for the control element drive mechanisms, considering dynamic interactions with the reactor vessel of a PWR, proposed a design approach to address intersystem LOCA, and presented the results of a feasibility study on the load follow operation without boron adjustments in the 1300 MW(e) KNGR.

The French poster presented the CIDEM project implemented by EdF to minimise power generation costs — an analytic systematic process aiming at the design optimisation with impact on investment, maintenance and availability. The project is oriented today towards the new French-German nuclear island of EPR. The paper focused on availability improvements and described how refuelling outages, turbine-generator overhaul and in-service inspection can be shortened by design provisions thus raising availability to values between 90% and 92% depending on the length of the fuel cycle.

A fourth category of posters addressed new reactor types including non electric applications of evolutionary water cooled reactors. There were five posters in this category from Kazakhstan, the Russian Federation and the Republic of Korea which presented more basic R&D aimed at the development of new evolutionary water cooled reactors with increased use of passive safety features and suitable also for heating purposes. The Russian and Kazakhstan posters covered a wide range of proposals, such as a Russian large PWR, its present status and prospects for the future, the application of a pool type reactor, RUTA, for district heating in fossil fuel deficient regions, and the application of NPPs for heating in Kasakhstan. One of the Korean papers presented a design concept for a light water cooled PHWR.

The other poster from the Republic of Korea, dealt with the development of an evolutionary passive PWR concept, the CP 1300. This concept is meant to be the successor of the KNGR and to be needed and built around 2015. The present status of the feasibility study points into the direction of a large passive PWR, also 1300 MW(e) range, including an increased number of fuel assemblies to reduce core power density, new core make-up tanks for smooth safety injection, decay heat removal by natural circulation in the steam generator secondary side, and passive containment cooling with internal or external condensers. The more detailed design, along with a comprehensive safety assessment including probabilistic safety analyses and economic assessment as well as validation tests for the new systems are planned for the future.

Two posters for Argentina and Canada dealt with specific aspects of proposed fuel cycles for evolutionary water cooled reactors. The poster from Argentina presented improvements in NPPs currently in operation, and their fuel cycle. The paper from Canada addressed the fuel cycle flexibility in CANDU reactors. The new CANFLEX fuel bundle with lower peak linear ratings and improvements in critical channel power and the use of slightly enriched uranium with lower fuel costs are briefly described. Particular attention is paid to the symbiosis of CANDU reactors with LWRs that provides the opportunity to directly recycle spent LWR fuel in a CANDU, either after conventional reprocessing and separation of depleted uranium or by using the DUPIC cycle, using only thermal mechanical processes to convert PWR fuel into CANDU fuel. Finally, the use of thorium is undergoing serious investigations.

The next package of posters dealt with I&C software and different structure aspects. Here we had by far most of the papers presented — at least ten — from Canada, Germany and the Republic of Korea. They dealt with instrumentation and control and software application as well as with developments in the infrastructure processes. The posters from the Republic of Korea dealt with the joint design and development of the KNGR man-machine interface, the development of an I&C system architecture for the KNGR, an advanced alarm system for the KNGR, the development of a digital safety system, called CPCS, using network and commercial operating systems, a design approach to computerise operating procedures for the KNGR and the development of the advanced I&C systems. Two Canadian posters addressed the use of simulation for CANDU I&C application at advanced processes for the production of heavy water.

A third Canadian poster dealt with software in safety applications and the fact that safety critical software needs a higher degree of confidence in its correctness than other categories of software. The poster described the practices used for real-time software engineering for CANDU NPPs and in particular the software development methodologies used for safety critical software engineering, namely the Rational Design Process and the Integrated Approach. Both methodologies have been implemented in the Wohlsong NPPs, and show a remarkably low number of errors.

The German poster gave a general view of the nuclear plant analyser ATLAS, developed by GRS for NPP safety analysis. The basis for the simulator is provided by best estimate codes for thermal-fluid dynamics and containment behaviour — among others the codes ATHLET and MELCOR. Up to now, the code has been used to simulate several German NPPs. Another simulation for a Russian NPP is in preparation. The code can be complemented by additional modules that provide the user with PSA based information or an analysis of emergency procedures using artificial intelligence techniques.

The last block of posters comprised five posters from the Republic of Korea, the Russian Federation, Slovakia and the USA, which addressed upgrading and backfitting of existing nuclear power plants as a precondition for deployment of evolutionary water cooled reactors, as well as advanced operating and regulatory approaches to be implemented in the future. The Korean posters dealt with the development of a new licensing system for evolutionary water cooled reactors in the Republic of Korea and the regulatory approach for computer technology applications to safety systems in Korean NPPs. The Russian poster described the development of nuclear energy in Russia and in particular the modernisation of operating and the construction of new NPPs. The Slovak poster addressed the safety improvement and results of commissioning of the Mochovce NPP of the type WWER 440/213. The USA poster finally dealt with the utilisation of risk informed in-service inspection to reduce O&M costs of NPPs.

Let me conclude with some personal remarks, after having read all these papers. All posters presented in this symposium confirmed impressively from the R&D, industrial, operational and regulatory side what has been stated in the invited papers, also. First, nuclear power, and in the more distant future, nuclear heat, present obvious environmental advantages and fit into the energy supply strategies of many countries in the world. Second, nuclear technology is still evolving and has the potential to cope with growing safety and environmental requirements using modern technological solutions in the fields of material sciences and informatics. Third, R&D, both in research organizations and industry, is active, task oriented and efficient and provides imaginative solutions to respond to the real needs of NPP operators. Finally, economic competitiveness and appropriate financing schemes are a serious challenge in the globalized deregulated electricity market, but nuclear technology has proved its ability to take it. Thank you.

OPENING OF THE PANEL DISCUSSION

P.E. Juhn
International Atomic Energy Agency,
Vienna

Thank you very much Mr. Kupitz and Mr. Foskolos for your excellent briefings.

Now, to begin the Panel Discussion, I would like to invite each panellist to provide remarks on some of the following topics:

- Will the objectives of sustainable energy and protection of the environment influence energy policies and the deployment of nuclear power?
- Will nuclear energy be utilized for purposes other than electricity production, e.g. process and district heating, and seawater desalination?
- Will the safety targets of the evolutionary designs be sufficient to gain public acceptance?
- Can new NPPs be economically competitive while meeting very stringent safety standards/ requirements?
- Why are so few evolutionary plants being built, in particular in the countries of origin?
- Is it reasonable to pursue several designs by several vendors, as the market is so limited?
- Can regulatory bodies, industry and utilities work together efficiently to achieve evolutionary improvements in the technology?
- What are the prospects for attractive financing schemes for evolutionary plants, in particular in less developed countries?
- Has the back-end of the nuclear fuel cycle been developed sufficiently for a wider use of nuclear power? Which elements need more attention? What are the prospects for wider use of MOX fuel in water-cooled reactors?
- What are the suggested roles for the IAEA in the topics covered by the Symposium?

PANEL DISCUSSION

Statement by D. Torgerson, Chalk River Laboratories, Atomic Energy of Canada Ltd, Canada

This list of questions was so interesting that I could not get past the first one. Therefore, I have decided to direct all my comments to the first question, which states "will the objectives of sustainable energy and protection of the environment influence energy policies in the deployment of nuclear power". I think this is a very critical question.

Over the past couple of days, I have heard about excellent technology, enhanced safety, solid efforts to reduce costs and what I see as a very clear vision of how nuclear reactors are evolving. It seems that we are doing almost everything right from the technical point of view. Nuclear power is getting faster, better, cheaper. Why is it then that nuclear power is not yet seen in some countries as a major part of the solution, for example, to global warming? Why in other countries is there a deliberate attempt to reduce the role of nuclear power?

Public policy is usually always aligned with public acceptance. However in the case of nuclear power it could be that the policy is being more influenced by vocal minorities than by the public. Whatever the reason, it is clear that we need to work much harder than we have in the past to replace perceptions with facts.

I once asked a senior engineer from a country that is attempting to shut down its nuclear programme, how they were going to replace nuclear power without harming the environment. He replied that the policy makers had the idea that if they could close the reactors down they would force the development of new technology such as solar and wind. However the reality is that these sources of energy are both dilute and intermittent, and are unlikely to ever provide sufficient baseload power for the growing demand. Nevertheless, the investment in these technologies has been substantial. For example, I read one recent study that showed that the USA has invested twice as much development funds in renewables than in nuclear energy. Yet, there is very little payback for that investment in terms of electricity production, and this situation will not change. By contrast, nuclear has been a very good investment.

The bottom line is that there are only three viable methods to generate electricity, now and for the future: fossil, hydro and nuclear. No other methods can produce the quantities of electricity needed. That is the key simple message that has to get across as the world begins to address the Kyoto accord. And when we look at the three methods of production it is clear that only nuclear power is environmentally benign. For fossil fuels the public understands very well that coal harms the environment. But I think that the natural gas producers have done a masterful job of positioning natural gas as a clean alternative. Even hydroelectric power has its environmental problems. The loss of land is an obvious issue. Also, recent studies in Canada at the Fresh Water Institute have shown that reservoirs that flood upland forests and peat soils release carbon dioxide and methane due to microbial interactions with the organic material. In fact, one such reservoir in Canada is reported to release as much greenhouse gases as an equivalent size coal burning plant. I think that's quite remarkable. It is not likely that many members of the public — or even policy makers for that matter — are aware that hydro electric power can be a significant contributor to greenhouse gas emissions. The inescapable conclusion is that nuclear power is the only way to significantly increase electricity production without affecting the environment.

Given this conclusion, I would like to consider the IAEA's role. I note that the IAEA's mandate is as follows:

> "The Agency shall seek to accelerate and enlarge the contribution of atomic energy to peace, health, and prosperity throughout the world."

The IAEA's mandate would appear to have the dual objectives of 'promotion' and safeguards. Here, of course, I am not using the term 'promotion' in the sense of advertising or lobbying. Rather, it means that the IAEA needs to continue to provide a balanced and factual view of nuclear technology. In addition, the Agency should continue to provide technology advice to countries considering the nuclear power option. In one sense, this 'promotion' is really part of the Non-Proliferation Treaty programme. Non-weapon states agree not to acquire weapons and to submit to full scope safeguards for the right to use nuclear technologies from other countries for peaceful purposes.

In conclusion, there is an urgent need to address global warming while at the same time planning to meet the increased energy demand in the next century. Therefore, it is essential that all knowledgeable experts and agencies do whatever they can to overcome the incorrect perceptions that might lead to policies that could restrict the use of nuclear power.

Thank you

Statement by W. Zeng, Guangdong Nuclear Power Company, China

I'm delighted to be here to share with you the viewpoint of a regional nuclear power producer of South China, the China Guangdong Nuclear Power Corporation, towards the future nuclear power development. My speech will mainly concentrate on the competitiveness of nuclear power, taking the Guangdong case as an example. However, before talking about our case in Guangdong, I would like to say a few words about this Symposium itself. I appreciate very much the effort made by our Symposium organizers, IAEA and KEPCO, to make this Symposium timely and open, and to conduct it so effectively. It gave the opportunity for the international nuclear community to put their representatives all together to review the situation, to exchange our views, and to discuss the challenges facing us, and finally to find a way to the future, with this very useful and also practical Symposium. I was a participant at the 17th World Energy Congress two months ago in Houston, USA. The atmosphere at that meeting was quite different from this one. People were speaking mainly about gas, oil and renewables. The voice of the nuclear power was very weak. But really it is a strong warning for us, actually.

It is a matter of fact that nuclear power is no longer in the 1970s. The world situation today is different from that time, when we started our study on evaluation of a water reactor programme. We need to adopt our strategies according to the current situation. Indeed, especially in the past five to ten years, the major new factor which influences on our nuclear power development is the deregulation of the power industry which is going on with a tremendous speed. In China starting from 1 January next year, two provinces and one city will start the testing period for the deregulation process. And other provinces will follow them very soon. Nuclear power has to compare with other sources of energy on a price basis. But at the same time the world's gas, oil, and coal costs have dropped down sharply. Even more the equipment and the consumption costs for fossil power plants have also decreased substantially. As a nuclear power producer, I would say all these facts make nuclear power difficult to survive.

As a utility we feel so strongly no competitiveness means no market. This is why we recommend to our nuclear power community, all parties involved should work jointly together and focus on the economic aspect. Give it as much importance as safety It is true, that no safety, means no nuclear power. But it is also true to say today: no competitiveness means no nuclear power. We are more than happy to see that during this Symposium more and more of our colleagues have recognised this point, and a lot of initiative proposals have been raised during this Symposium especially on how to balance the safety and costs, and how to improve our technical solutions in order to keep the safety level, and at the same time, reduce costs. We hope after the symposium a meeting could be organized to bring these creative ideas into a realistic recommendation.

I would like to come to the Guangdong case study. Guandong is one of the fastest developing areas in China. Starting in the 1990s, the gross domestic product (GDP) and the power generation in Guangdong province increased by 18,5 to 18,6 percent respectively every year. The price for such sharp increases in the past decades is environmental pollution especially for the delta area of the Pyr river with acid rain due to too many coal fire power stations there Seventy five percent of the power generation relies on coal. Second there is an unbalanced energy structure there. Especially there are to many diesels and small size generators due to the rapid industrialization to satisfy the growth of the economy. So we have in total about 11 Giga watt of diesels and small size generators, by which I mean 50 Mega watt generators. Of course it creates a lot of trouble in the current situation. The condition for Guangdong for the future is sustainable power economical development, during the next ten years. In1997 we had a total installed capacity of 28.4 Giga watt. But according to the planning of the Guangdong government, by 2010 we need to add another 30 Giga watt. According to the Guangdong government planning, and subject to the central government approval, we are planning to build another 6000 Mega watt nuclear power in the Guangdong area before 2010.

Why — because we think that from a strategic point of view, to keep sustainable development in Guangdong area, the preconditions for such kind of substantial level power development is security of the supply, environmental protection, and restructuring of the power structure. So we have selected nuclear power primarily due to the same reasons as Japan, and the Republic of Korea — because in Guangdong we lack all the primary energy sources. We have no oil, no hydro and very little coal- only about 20% of the total needed supply. So, by the way we have built up the Daya Bay and Lingao power stations. We have accumulated a lot of experience, giving us the foundation for the future development. So the Guangdong government has decided to develop nuclear power as an alternative energy source.

In Guangdong in 1996 the coal and gas fired capacity was in total about 71%. The hydro, wind and other pumped storage was only 21%. Nuclear was only 8% of the total. Apparently the structure is unbalanced in that there is too much reliance on coal fired power stations and these cause a lot of pollution in the area and a great pressure on the transportation system, also. So the intention for the Guangdong government is to restructure. By 2010 we intend to have the coal power at about 50 percent and nuclear will occupy another 15 percent of the total installed capacity, even though this is not sustainable in the future development. So the final aim for the Guangdong government is to reduce coal fired power station fraction down to 1/3 of the total and nuclear another 1/3, and others another 1/3. We think in such case we will keep the sustainable development in Guangdong.

Now I wish to discuss my second topic. The consideration between the plant betterment and the installed cost per kilowatt is necessary. Yes, as I mentioned in my introduction, I think as a utility the more advanced the better, the more safer the better. But at the same time we need to look at the economic aspect. Especially we need to take care because the deregulation process is coming very soon. And during our feasibility study this is the major concern, because when we discuss with the electricity company the only one thing is what are the cost. So, yes, we understand at the moment we have the market, but whether we can occupy it is quite depending on the competitiveness. So when we set up the strategy for our project for the next Guangdong Units to meet our plan in the feasibility study to build another 6000 Mega watt before 2010, the major concern for us is how to make the nuclear power comparable at least with the clean coal fired power station. This is essential.

So, at the moment we are negotiating with all the existing vendors who can provide a proven design with proven technology, with the reference power station to see whether we can, on the one hand, come as close as possible to URD and other requirements, and, at the same time, to keep economic competitiveness. So as a transitional solution we are thinking we may or may not adopt the full requirement of the URD. It means we will negotiate with all the vendors to see how to upgrade the classical designs to be as modern as possible while at the same time having a reasonable price. This is our solution.

Of course as a utility we hope very much that for the long term, for example by 2010, that the next generation of reactors will meet both goals: safer and cheaper. In that case we will adopt our policy and buy the more advanced designs. So for us we think in order to reduce the construction costs per kilo watt is essential. Here are some recommendations: One is optimisation of the unit power level. Second is to maximise the localisation of the manufacturing and design with the aim to lower the total cost. It is our experience in our project that when we adopted the localisation policy and tried to implement it, we go a lot of benefit from the difference of the labour cost and also from the difference in the material cost. So we need to seriously study how to jointly proceed in this direction. That is why we have put the self reliance programme in our Guangdong project 3. And, of course, it is important to focus on methods of increasing the plant availability. We intend to have availabilities of 85 to 87 percent — the higher the better. And it is important to examine ways to minimise the operation costs. We feel we can meet the competition on this basis.

Next I will come to the prospect of attractive financing schemes. Financing is always is a very difficult issue for the developing countries because nuclear projects are capital intensive. In our case the financing costs could be about 15 to 20% of the total cost. So it is quite essential. It is why we always have to discuss with our potential supplier to see how to achieve an attractive financing scheme. The key areas for Guangdong that we are trying to approach are, first, whether the project financing is possible, and, second, a traditional area, whether there is government support to get the export treaty. Both areas are certainly very difficult to get the nuclear power project in the project financing. Right now we are working with vendors, and suppliers, to see how we will make it happen. Finally, of course the involvement of the local bank to finance the local work force is very essential.

Finally, I would like to take this opportunity to emphasise some existing facts, which we think are unfair to the nuclear power as a clean energy. Number one is the OECD still applies an additional one percent interest rate on nuclear projects compared with fossil projects. We think that nuclear power should not be penalised, but should contribute as a clean energy source in the future. Second, the World Bank and the Asian Bank are not yet open to nuclear projects. Thirdly, some countries provide favourable overseas loans for environmental protection, but not for nuclear projects. Whatever the reasons for these policies, we believe it is time for the international organizations to review their policy towards nuclear development, at least on a case by case basis.

Thank you very much for your attention.

Statement by A. Birkhofer, Gesellschaft für Anlagen- und Reaktorsicherheit, and Chairman of IAEA's International Nuclear Safety Advisory Group (INSAG)

Thank you, Mr. Chairman. I can be very brief because many things have been stated here with which I agree completely. I would like to start with some remarks concerning the world-wide acceptance of nuclear power and the interfaces between nuclear power and other energy technologies, e.g. the consistency of the overall approach. I believe this is one of the most important issues where changes are necessary. I attended some G7/G8 meetings on energy in 1998, and I remember that it was very hard, even with the ministers of economy, to achieve agreement on an overall approach to energy including nuclear power. This is a problem we have to address systematically. We must try to get this issue to a political level capable of promoting the idea of integration.

Second, I would have liked to have a conference like this under the co-sponsorship of the OECD and the IAEA. Why? Because even though the OECD does not cover all of the countries involved here, it has a very broad range of activities extending far beyond nuclear issues, and it is far from being considered biased and promoting nuclear power. Therefore the IAEA should use the framework of the OECD more frequently for organising meetings like this one. In 1997 I had the pleasure to review the role of the OECD Nuclear Energy Agency for the Secretary General of the OECD. I realised the tremendous need for an improved cooperation, especially between the OECD/NEA, the IEA, the OECD itself and the IAEA. Many players have to work together and with other sectors in order to make nuclear power issues well understood not only by those involved in day-to-day work. We have to work for that.

Technical competence is also a significant issue, and I would like to refer to Europe and the USA in that regard — I'm sure Mr. Magwood would agree. The use of nuclear power in these two regions of the world is stagnant. Important reasons for this situation are the over-capacity of electricity generation in some areas and the current low price of gas. During this period of stagnation, which might continue for another ten or even fifteen years, we have to maintain our technical competence in order to guarantee the safe operation of the nuclear power plants. This is crucial for the OECD countries which encompass about 80% of all nuclear power plants in the world. But it is also a big challenge for the IAEA which is involved on a even wider level.

Another important topic is the relationship between economics and safety under the current conditions. Under the increasing pressure of competition in deregulated electricity markets cost reductions sometimes receive so much attention that safety is in danger of being neglected. I think it is important to recognise and I would like to make a very strong commitment that a reduction of safety does not necessarily lead to a reduction of costs. Several examples show that the reduction of costs is possible while even increasing safety.

Let me reflect very briefly on the historical background of some current key efforts for improving safety. After the Chernobyl accident, the INSAG formulated basic safety principles for nuclear power plants (INSAG-3), describing the foundations of a good safety practice. By the way, INSAG will soon give out a revision of this book, taking into account more specifically the new developments of nuclear power in some of the countries. Then, in 1991, we had the very important IAEA conference "The Safety of Nuclear Power: Strategy for the Future". We debated enormously on what had to be done. The questions were more or less the ones we discussed here, although maybe less sharp concerning the economic situation. And we all agreed that improved safety is a prerequisite for new nuclear power plants. We discussed the role of the containment and made recommendations for strengthening the containment function in order to effectively mitigate severe accidents. This opened the way for improved system configurations with more safety at lower costs. A good example for an improved design resulting from these recommendations is the European Pressurised Water Reactor.

Regarding the question as to why so few new nuclear plants have been built particularly in the countries which originally developed the nuclear technology, we have to realise — as mentioned earlier — that most of these countries have a saturated capacity for generating electricity. Regarding the question on the suggested role for the IAEA, certainly the most important point for me is achieving a broad consensus on issues of general significance. A very high priority issue is the disposal of nuclear waste on which we have to build up a more stringent consensus. And we have to make more of an effort and make that effort more visible to the world community. Another important subject is low level radiation. I think it is necessary to compare it generally with background radiation, because people understand this much better. We have to resolve the issue of low level radiation and not have it as a dispute forever among the experts of different kinds. Another important issue is the role of nuclear power for a sustainable energy supply. We should make more visible how nuclear energy can contribute to sustainability as Mr. Torgerson already stated. This is a field where the OECD with its involvement in many areas concerned should contribute and will contribute according to the words of its Secretary-General. And there is certainly the very important role which we discussed here, building a consensus on getting the forum for concepts and principles. Thank you very much.

Statement by A.K. Anand, Bhabha Atomic Research Center, India

Thank you Mr. Chairman. One of the earliest IAEA meetings on ALWRs was held in 1986 in Washington D.C. which I attended. In this meeting a number of new idea were discussed. It was pointed out that the power reactors were becoming bigger and more sophisticated and thus were not suitable for smaller grids and for developing countries. The future reactors should be smaller (= 600 MW(e)) and should be simpler. The ideas expressed were to scale up the simple and robust propulsion reactors. The evolutionary reactors discussed were AP-600 and SBWR and revolutionary and forgiving reactors discussed were ISER and PIUS. A number of ideas of ship mounted power plants were presented by Japan. The idea was to use the much developed ship building industry, Now in 1998 all but AP-600 have been dropped and not even one plant has been built.

Dr. Hatcher mentioned about the cost reduction, mass production and mentioned about Henry Ford. If we look at the period of second World War a number of ship yards in Germany, USA and USSR were producing submarines at rate of 1 every two weeks or so. Each submarine was a complicated piece of machinery weighing a few thousand tons. If this could be done about 50 years back, the reactors can also be produced for electricity generation at the same rate. Nuclear Steam generating plants can be produced in some yards and Turbine plants and C&1 panels in some other plants. Civil works can be prepared to receive the plants. At the plant site there should be no need to make more than about 100 welds and maybe a few hundred cable and wire connections. If we start working now this can be achieved for our future generations in 2 to 3 decades. These plants can be used for electricity production and / or desalination of seawater.

Statement by Y. Oka, University of Tokyo, Japan

Thank you Mr. Chairman. Today I want to comment on economic conditions of nuclear power, and on some technical innovations.

Economic deregulation: Economic deregulation has become a global phenomenon. The electricity supply industry is undergoing rapid reform throughout the world. Its traditional attributes such as monopoly status, extensive regulation, and a high degree of public ownership are being replaced by more complex ones. In many countries competition is enforced in the electricity supply industry.

New construction of nuclear power plants is getting into difficulty after deregulation in some countries. Small investment is preferred under cloudy circumstances as now. Combined cycle gas turbine plants are selected because of their small capital cost and short construction period. Nuclear power is capital intensive. Its option value is low in some countries at present.

But definitely nuclear power has great advantage in preventing global warming. Nuclear power is very competitive after depreciation of the initial investment.

Economic characteristics: Let's examine economic characteristics and investment risk of nuclear power. The capital cost of nuclear power is high. Initial investment is very large. It takes several years before the electricity sales can begin. These characteristics do not fit well to the small company entering the electricity generation business. This problem is very tough for nuclear power, because its cost reduction has been mainly achieved by utilizing the economy of scale.

Another characteristic is the uncertainly in future costs. Investors cannot be convinced well with the construction and operation schedule of nuclear power unless societal regulation is stable. The cost of disposing radioactive waste depends on the solution of the problem. Nuclear power needs long term investment and successful operation. A time lag between cash inflow and outflow exists due to the expenditures for radioactive waste after shutdown. Nuclear power requires a national commitment to R&D, risk regulation, non-proliferation and disposal of radioactive waste.

Japanese circumstances: Fortunately until now, some of the characteristic have not shaped Japanese nuclear power program. We, Japan dose not have energy resources, importing almost all oil, natural gas and coal. Fossil fuels are very expensive. Generally speaking, nuclear power is competitive with fossil-fired power plants. But electricity generation was deregulated. Economic competition with Independent Power Producers (IPPs) has come in the scene. The potential IPP capacity is large, 20 GW(e), which is half of the current nuclear power capacity. The competition with fossil fired power will be influential in the near future. Japan also has a unique position in meeting the Kyoto protocol through reduction of emitting greenhouse gas. Energy conservation measures have already been implemented in many areas, because of the high fuel prices. No large opportunities remain for further conservation. Without adding 20 more nuclear power plants, it will be difficult to meet he Kyoto protocol. Large sized evolutionary LWRs are the next generation LWRs in Japan.

Competition with fossil-fired power: Recently, advances in combined cycle gas turbine (CCGT) plants are remarkable. They achieved high thermal efficiency and high power rating per unit. Capacity can be added in a modular way. The construction period is short. In addition, operation and maintenance costs are low. It is based on the jet engine technology for airlines, always cautious about O&M.

Fossil-fuel resources are abundant in the next century. Natural gas has become available from deep sea beds, such as Gulf of Mexico, because of the drilling technology innovation. There are similar geologic formations in the world pouring the big river to a Gulf. The estimated reserve of natural gas is 84 years. The increasing competition among power vendors has exerted pressure on power plant prices. The prices have practically been halved only in a few years. Distributed power generation will be popular in the next century. Small-turbines, diesels are being implemented for application to hospitals, hotels and office buildings, and fuel cells are under serious development.

Cost reduction of operating reactors: Let's examine the cost reduction strategies of operating reactors. Increase in the capacity factor is the key element for cost reduction of operating reactors. It will be improved by longer cycle, reducing unplanned shutdowns and shorter refueling outage. For example, the shifting from periodic to failure oriented valve maintenance has potential cost savings. Increasing the fuel burn-up reduces the fuel cycle cost. Staffing and contracted services are reduced successfully in some utilities in the world.

Enhancing the rated power by clarifying the design margins improves the cost without any hardware changes. Pooling emergency spare parts such as large motors and pumps among utilities reduces the investment, warehousing space and inventory, and therefore the cost to each utility.

The capacity factors are improving after the deregulation in some countries. Also safety performances are improving. Cost reduction is compatible with safety improvement. Keeping the high level of safety and minimizing incidents are foremost important for economy of the operating reactors because an accident is very costly. So the safety is first and cost is the second for operating reactors.

Organizational issues are one of the safety concerns of nuclear power, but we already had the solution. The productivity enhancement and quality control and quality assurance are the keys to solve the problem, when implemented correctly. This may be the reason why capacity factor is improving with reducing staffing in UK and USA. Sharing the experience and warm encouragement rather than punishment are the way to go.

Cost reduction strategies of advanced LWRs: After the depreciation of the initial investment, nuclear power is very competitive. But new construction is important for the bright future of nuclear power. Cost reduction is the major issue of advanced reactors. The goal of cost reduction is to compete with advanced fossil-fired power plants. This is a tough challenge, considering their improvements.

Increasing unit power rating has been the most effective way of reducing power cost. But it is not compatible with the current market trend of decreasing the investment volume. It will be difficult to find a cost-competitive solution for nuclear power by decreasing unit power rating. Constructing modular reactor units could be a solution. However, multi-units siting is already popular in Japan, and almost all are large-sized units. The power rating of the next generation LWR will be in the range of 1500 and 1700 MW(e).

Shortening the construction period is effective in reducing cost. ABWR construction at Kashiwazaki was completed in 51.5 months.

Standardization of the plants is a well-known element of cost reduction. Modular fabrication of components and modular construction of the plants are also effective.

Technical development is evolutionary. The final product will not suddenly appear. Improving technology so as to back fit to the existing reactors will be a good strategy of nuclear power cost reduction and technical developments.

The strategy of reducing O&M and fuel cycle cost can be learned from the operating reactors experience.

Socio-political issues are sometimes more influential on economy than technical ones. It is necessary to correctly address these issues. Socio-political issues cannot be solved technically in many cases. The fear of radiation is a source of anti-nuclear feeling of general public. The linear non-threshold theory of radiation exposure is too conservative in the low-dose range, I think. Not to over-respond to accidents and incidents is foremost important.

Lastly the extreme strategy of cost reduction is to seek technology innovation. This is the subject what I want to talk a little more today. I am from academia, rather easy to talk.

The dynamics of innovation of the manufacturing industry, is described in the book entitled "Mastering the dynamics of innovation" by Professor Utterback of Harvard Business School. Product innovation dominates at first. After the dominant product design, the design holding the largest market share is established. Production process innovation follows. LWR is the dominant product design of nuclear power. I think LWR is in the era of production process innovation. Standardization is also a type of production process innovation.

The modular construction processes used at the Kashiwazaki-Kariwa ABWR are a type of production process innovation. A construction period of 51.5 months was achieved. Use of computer technology such as CAD is also the development for this direction. I think further innovation of production process will be one of the ways to go for cost reduction of LWRs.

Let us examine the historical competition among thermal engines. At present it looks that the steam powered, internal combustion engine, jet engine and rocket engine share their applications. Steam power is used mainly for large central power stations, internal combustion engines are used for automobile, ship etc., jet engines for aircraft and rocket engines for rockets. But historically it was not. Steam power was used for automobiles in the middle of 19th century. But when oil-refining technology was established, it gave the way to internal combustion engine. The replacement of steam power with diesel engines for ship propulsion occurred very gradually from 1910 and 1960, from small-power to large-power. It took 50 years for the replacement. Now on the sea, steam power is used only for the special application such as nuclear-powered icebreakers, submarines and aircraft carriers.

Jet engines are now invading the main territory of steam power as advanced combined cycle plants. These changes occurs very slowly, sometimes it takes 50 years. The reason is that the technology evolution essentially depends on the current applications and its experience. We sometimes do not notice the change, but it will be a mistake to neglect the needs of innovation.

Now lets examine the historical evolution of boilers The primitive boiler is like a large "kettle". The heat transfer is enhanced in the circular boiler by immersing the heat transfer surface. Water tube boilers have been evolved by enhancing the coolant circulation. The capacity has been increased with the evolution. The once-through boiler is the most advanced type of boiler. The operating pressure is supercritical. The LWR is a type of circular boiler with forced circulation of coolant.

I have been developing the concept of supercritical-pressure light water cooled reactors for 10 years. It is a reactor version of the once-through boiler.

The critical pressure of water is 22.1 MPa, which is about 40% higher than that of a PWR. Supercritical water does not exhibit a change of phase. Steam separation and coolant re-circulation of water are not necessary. All feedwater can be fed to the turbine. The plant system is once-through, direct cycle like the supercritical fossil-fired power plant. The separator, dryer and re-circulation system are not necessary. The reactor vessel becomes small and the control rods can be mounted on the top of the reactor. The reactor vessel and control rods are similar to those for a PWR, the containment & safety system are similar to a BWR, and the BOP is to similar to that for a supercritical fossil-fired plant.

The coolant enthalpy inside the containment is less than one fourth of a ABWR. The position of the reactor vessel can be lowered because of the top-mounted control rods. This is an advantage in seismic design and an advantage for the construction period. The main coolant pipings are only two.

The thermal efficiency of a supercritical-pressure light water reactor is approximately 44%, higher than that of the fossil-fired plants because of no chimney loss. The coolant flow rate is approximately one tenth of an LWR, but higher than that of fossil-fired plants. The main steam is consumed for re-heating in nuclear reactors, but it can be returned to the furnace and heated again in the fossil-fired plants.

Light water reactors were developed 40 years ago. The success was based on the experience of sub-critical fossil-fired boiler technologies at that time. Supercritical boilers were developed 40 years ago in the USA. In Japan the first supercritical boiler, Anegasaki Unit 1, started operation in 1967. Since then they have been constantly improved, because of the high fossil-fuel prices. The sliding-pressure plant was developed in 1980. It can operate at sub-critical pressure to follow load change. Currently higher pressure plant and higher temperature plant are put into operation. The operating experience of supercritical boilers out number the LWR, approximately 400 units in USA and 150 units in Japan. Considering the evolution history of boilers and this experience, supercritical-pressure light water cooled reactor will be the way for the innovation. Of course it is not an easy way, but potential cost reduction will be great. The goal of the cost reduction is 50% from ABWR.

In summary:

Safety is No.1, cost reduction is No.2 for operating reactors, of course both are important. Economic competitiveness is important for advanced reactors. Production process innovation will be the source of cost reduction. Innovation of technology is important in the long term. Seek innovations..

Thank you for the attention.

Statement by Chang-Sun Kang, Seoul National University, Republic of Korea

Recently, the Korean Government has been investigating what should be done in the domestic nuclear power industries to strengthen their competitiveness. Pursuant to this government move, I had the lucky chance of conducting a study called "Nuclear Vision Towards the 21st Century in Korea, and our Missions". In this study I reviewed the roles played by nuclear power during the recent economic recession in the Republic of Korea, the so-called IMF crisis and forecast the future prospects of nuclear power. I would like to briefly introduce the general outcome of the study which some of you might be very interested in.

There are four major roles of nuclear power identified in the study. The first role is that *nuclear has assured the stable supply of energy.* In 1997 we imported more than 97% of energy required from foreign countries. Fossil fuel imports were 87% and oil dependence was 63%. This is a very weak and fragile structure of national energy supply in any sense. Nuclear power facilitated the stable supply of energy because of its high fuel stockpiling capability, by nature, and it also substantially reduced the fuel — especially oil-dependency.

The second role is that *nuclear has reduced the national trade deficit.* The cumulative energy import for the last ten years starting 1988 through 1997 was 15.2 billion US$, while the total trade deficit during the same period is only 4.4 billion dollars. Therefore, we can easily see the energy import has been the main contributor to the present huge trade deficit in the Republic of Korea which can be substantially reduced by nuclear power.

The third role is that *nuclear has lowered the electricity bill.* In 1997 the electricity generation cost by nuclear was about 23.7 mills/kWh (expressed in US$) which was the lowest among five different generating choices, and the portion of fuel costs was 10.2% which was also the lowest among them. The low generation of costs of nuclear accelerates the growth of our economy and the low portion of fuel costs enables the achievement of our energy self reliance.

The fourth and last role is that *nuclear has mitigated the global warming problem.* Nuclear power plants only produce 8 grams of CO_2 per kW·h of electricity generation, while coal fired plants produce 270 from fuel burning and 24 grams per kilowatt from non-fuel operation. Therefore nuclear is the only viable solution to reduce CO_2 emission coping with the UN climate change convention.

Based upon the four roles that I described above, there are two major missions and then one vision toward the 21st century.

Our *first mission* is to *establish nuclear as the major source of domestic electric power production and seek for a stable supply of energy*. We can accomplish the mission specifically by increasing the nuclear mix to higher than 60% in the national electric supply frame; by developing hydrogen and electric cars along with producing hydrogen and/or charging batteries, utilising night time access electricity generated by nuclear power, which would fundamentally eliminate our high oil dependency problem; and by extending the use of nuclear energy at an enlarged scale beyond electricity generation, such as in district heating, sea water desalination, ship propulsion and hydrogen production.

The *second mission* is to *achieve the self reliance of nuclear reactor and fuel cycle technologies*. This mission could be achieved by improving the economics of the Korean Standard Nuclear Power Plant; by developing the KNGR and the multi-purpose reactor SMART; and by performing R&D work on KALIMER which is a 300 MW(e) liquid metal reactor. For nuclear fuel cycle technology our mission could be achieved by developing advanced fuels like high burn up and MOX fuel, advanced CANDU fuels like CANFLEX, and tandem fuel like DUPIC.

The *vision towards the 21st century* is to *develop nuclear technology as a profitable export product in due course of performing our missions*. In this study I had a chance of reviewing the present situation of world electric utilities. First of all I found out that there exists a large excessive supply of electricity in Europe as well as in North America. No more electricity generating units are required to for a while — whether it is conventional or nuclear. In ten years, however, the revival of the electricity market has to come up due to replacement of old and inefficient units. Nevertheless newly introduced units will not be fossil power plants since the European Union committed to 8% percent CO_2 reduction, and the USA 7%. There will be no choice other than nuclear power generation. A huge world nuclear market is expected by 2010, and the nuclear power industry should be prepared for it. Therefore, until then I think the survival is the key issue world wide.

In coping with the present situation, world nuclear industries are going through a big reform. Big merging acquisitions are going on, like ABB and CE, Framatome and B&W, and BNFL and Westinghouse. Governments are still trying to hold the grip on controlling nuclear industry since they consider nuclear is too important to be left completely to private organizations. For example, the Canadian parliament supported AECL with about 160 million dollars in 1997. In spite of a strong privatisation policy of the UK, the Department of Industry (DTI) still owns 100% of BNFL shares, and allocated MAGNOX plants for its electricity generation business. In France, the government still controls 51% of Framatome shares.

Meanwhile the internal reform of business is still in progress: by maintaining minimum engineering capabilities rather than huge manufacturing facilities, by concentrating efforts and nuclear services and fuel supply for operating plants, and by diversifying business such as decommissioning , radwaste management, etc.

In the next ten years through a big reform in my opinion, nuclear power industries will be restructured based upon advanced reactor concepts, most probably, various evolutionary types of water cooled reactors, for example the EPR in Europe by Framatome and Siemens, and AP-600 of BNFL with Westinghouse, ABWR of General Electric, Toshiba and Hitachi, CANDU-9 with AECL, and KNGR.

I will briefly talk about the status of nuclear technical and business capabilities in the Republic of Korea. You can read about our technical capabilities in many papers and posters presented in this Symposium, so I'm talking now only about the business side. Our business situations are a lot brighter. We are providing two 1000 megawatt size KSNPPs to North Korea through KEDO. I think it is the beginning of our export activities. In the fourth electric supply development plan which was announced last August by the Korean Government, 18 additional units of nuclear power plants will be constructed by 2015, in which KNGRs are definitely included for timely completion by 2010. In summary, the Republic of Korea is in much better situation than any other nation. the Republic of Korea guarantees at least one reactor construction annually by 2015 and maybe expects more from the North.

I'd like to conclude in this way:

- The role of nuclear power has been well recognised in the Republic of Korea especially during the recent IMF crisis. It is imperative to sustain a strong nuclear power programme in the Republic of Korea.
- The Republic of Korea shall accomplish self reliance of reactor and fuel cycle technology through comprehensive and systematic R&D programmes.
- In due course, the Republic of Korea will achieve economic competitiveness in the world market and develop the nuclear power technology as a profitable export product on time.
- The Republic of Korea has to get ready for all this including restructuring the industry, which is by the way ongoing now — our industry is going through a big reform of restructuring, privatisation, merging, acquisition and commercialisation. Our nuclear industry is not an exception.

Thank you very much for your kind attention.

Statement by W.D. Magwood IV, Office of Nuclear Energy, Science and Technology, US Department of Energy

Thank you Mr. Chairman.

I first would like to take a moment to thank our hosts KEPCO and the Korean people for this conference. This has been my first trip to the Republic of Korea — I have been very impressed with Seoul, in general, but also with the technical capabilities of the people I met at KAERI yesterday. This trip has confirmed everything good I've heard about the Republic of Korea and its nuclear expertise. So I would like to congratulate the Koreans on this.

During my trip I have had the opportunity to walk around a bit. Yesterday while waiting for a train in Taejon I visited a market. While walking through the market I came across several ladies who were selling vegetables and prawns and other things. Looking at what they were selling I noticed they tend to specialise. One lady would sell fruit, another fish. They didn't mix and match too much. They specialised. And I couldn't help but wonder, did one lady sell fruit because she liked fruit? Did one sell prawns because she liked prawns?

I think it's pretty clear that whether they liked fruit was irrelevant. What was relevant was what people were willing to buy from them and what they thought they could sell, especially with all the other competition they had around them.

I won't pretend to speak for all countries, but in the USA I believe the utility chief executive officers (CEOs) think a lot like these ladies in fruit shops. They are extremely pragmatic and don't purchase or sell what they don't think they can make money from.

With this in mind, I will respond to the question of why there hasn't been more construction of evolutionary plants in their countries of origin. This question is pointed more at the USA than anyone else. The answer from the US perspective is quite simply is because we don't need it.

As several speakers have already pointed out, the USA has excess capacity right now. That is shrinking, but we still have excess capacity. We also have many resources: coal, which we are continuing to burn; natural gas, which is becoming increasingly popular; and we also have renewable resources that are viable in different parts of the country. In California, for example, geothermal is viable. In some parts of the country wind power is starting to show promise. In other parts, wood chip burning co-generation is an option. There are lots of options and we are exploring all of them.

In addition, and probably most important in the USA and I believe in Europe as well, the use of more efficient technologies is more of a competitor to new generation than even natural gas. If you talk to utility management personnel, they will tell you very bluntly that they can make a lot more money by encouraging their customers to become more efficient than they can by building any particular type of generation. So efficiency also has had a big impact on nuclear power because it has helped the utilities moderate growth and power needs. In the USA, the projection over the next 20 years or so is that electricity growth will increase by about 1.5% per year. I think Europe is probably pretty close to that. Those facts alone explain why we're not looking to build a lot of new capacity in general and nuclear capacity in particular.

That situation will start to change in the out years because we will need to deal with the climate change issue. In the near term, efficiency, more application of renewables, more efficient electricity generation with fossil fuel, and increased use of natural gas will all occur. The US Government's official projections of what fuels will be used to generate electricity in the USA show that the nuclear component will begin to decrease around 2010. Natural gas is projected to skyrocket, and the use of renewables to increase somewhat. I think that one of the most important things about such projections is that they are almost certainly wrong. It is easy to get excited by the prediction that nuclear is likely to decrease, but I know on what these predictions are based and I know that these assumptions will need to change with time. It is not unusual for governments' projections of which energy sources we will use to prove wrong. This is the important thing to remember when it comes to nuclear power. Understanding that government projections can change, and understanding that the public does not care which energy source is used so long as electricity is available, leads you to conclude that the best course is to leave the decision to use nuclear power in the hands of the private sector. The private sector — while not infallible — over time has made the right decisions. That's what's going to happen in the future.

Nuclear power as a method of generating electricity in the USA is extraordinarily efficient and cost-effective. There is almost no better way of generating electricity from a cost standpoint than using the existing 105 nuclear power plants. The restructuring that the electric industry in the USA is going through right now is going to prove that this efficiency is a real advantage for the utilities that operate these plants. Perhaps I should rephrase that because it's entirely possible that some of the utilities that own operating nuclear power plants are having financial difficulties because of the initial capital cost of the plants. But the fact is that the plants that are very efficient will continue to operate. I think the vast majority of US plants will receive license extensions and will still be operating close to the year 2050.

On top of this we have the climate change issue. The first question asks if the objectives of sustainable energy and protection of the environment will influence energy policy and deployment of nuclear power. It is very difficult to determine how the climate change issue is going to impact nuclear power. I think the impact that these issues have on nuclear power will depend entirely on how much the public and the governments of the world really try to deal with the problems. It's easy to say that we all should reduce CO_2 emissions but that's a lot harder to do than it is to say. In the USA it may prove difficult to reduce CO_2 emissions to anywhere near the Kyoto levels without some clear governmental action to encourage the use of certain energy options and to discourage the use of others.

When you ask utilities how they are going to deal with meeting the voluntary portion of the United State's climate change greenhouse gas reduction programme, it's pretty clear that they plan to use a lot of nuclear power to do that. Most of the contribution towards limiting CO_2 emissions over the last 20 years has come from nuclear. Nuclear power will remain an important element in any reasonable strategy to reduce CO_2 emissions. The job of this industry and those of us in it is not so much to worry about when the next plant gets built as it is to maintain an infrastructure and a nuclear intellectual capability that we can use when the time arrives, because that time will arrive.

This may be a very strong statement but in my view, continued and even expanded use of nuclear power is inevitable. I think that — given the limited resources, the impact on the environment, the success that we really have had in nuclear power that over an extended period of time — nuclear power may even become perhaps a dominant form of electricity generation over the long-term. Now that's a strong statement, but it's my opinion.

That said, there are some things that need to be done. Primarily speaking for the US Government, dealing with nuclear waste issues is probably the single most important issue for nuclear power. Giving utilities certainty that nuclear waste will be dealt with efficiently and cost-effectively is absolutely critical to the future of nuclear power. If we don't deal with that successfully in the next decade, it will make current plants seem less of a good deal for utilities, which in the longer term will have an impact. Next, one of the issues that I have begun to think about more carefully as a result of this Symposium — because so many speakers brought it up, including S. Hatcher who presented an extremely good paper on the first day of this conference — is how to reduce construction costs. This may prove to be the most important issue for us to deal with. While other technologies, such as gas-cooled reactors and liquid metal reactors, have their advantages, reducing construction costs of whichever technology we bring to bear will prove to be the most important issue. One of the things I plan to do when I get back to Washington, D.C. is to think about how to work with other countries to deal with the construction cost issue. The advanced light water reactor programme was really a major success for us. But one of the factors we have to recognise — given that we aren't building plants now and that the earliest that the USA is likely to build plants is 10 to 15 years in the future — is that the ALWR programme as completed may not be enough. We may need to go beyond that, and going beyond that may mean designing reactors that have passive safety systems but also are vastly less expensive to construct than our current designs.

There are two other jobs we have. One is to preserve the intellectual infrastructure. The US Government has started to put programmes in place to preserve the university infrastructure, to try and preserve the intellectual capabilities of our national laboratories. We will do more in that area. A second area that needs to be addressed, and one of the other speakers brought this up also, is the need to level the financial playing field for nuclear power. This is an area of government policy that really needs to be addressed and dealt with.

So, in conclusion, dealing with nuclear waste, reducing construction costs, preserving the intellectual infrastructure, and levelling the financial playing field should be the four highest priorities for us. While technology work is important and we ought to continue to research and develop advanced reactor designs, dealing with these more pragmatic issues are going to be more important in the future than anything else.

Thank you very much.

CLOSING REMARKS

P.E. Juhn
International Atomic Energy Agency,
Vienna

It is my honour to make closing remarks on behalf of Dr. ElBaradei, Director General of the IAEA.

More than 350 participants from 32 countries and four international organizations attended this Symposium. This Symposium has provided a forum for information exchange on technology advancements and the readiness of evolutionary water cooled reactors to contribute to the near and medium term energy needs. Strategic issues, technologies and economic viability of evolutionary water cooled reactors have been addressed. Invited papers have been presented on important subjects by internationally known and high level persons. Many of the papers have been of multiple authorship from different countries or international organizations, and reflect significant international co-operation by the authors in preparing their contributions. With the publication of the Symposium Proceedings, the IAEA will provide all Member States with a balanced and objective source of information on the key issues, and on the technology advances which are being incorporated into evolutionary water cooled reactor designs. The IAEA highly appreciates the efforts of all authors and speakers in preparing and presenting their papers and posters at this Symposium.

We sincerely hope that all participants have found these three days to be of high interest, and will benefit in the future from the information and discussions of this Symposium.

Finally, I would like to thank the Government of the Republic of Korea and the Minister of Science and Technology, His Excellency Minister C.H. Kang for inviting the IAEA to hold this Symposium in Seoul. This is my home country, so I also personally wish to express sincere appreciation and gratitude to the Republic of Korea and to Minister Kang for hosting this meeting. I would like to thank the President and Chief Executive Officer of the Korea Electric Power Corporation, Mr. Y.S. Jang for his role as host, for his generous hospitality, for his efforts as Symposium Chairman, and for providing funds and manpower from KEPCO for the Symposium. We are also grateful for the kind support of the industrial organizations which have presented their designs, and for the co-operation of the OECD Nuclear Energy Agency, the Uranium Institute, the Korean Nuclear Society and the Korea Atomic Industrial Forum.

I would also like to thank the Symposium Steering Committee, and staff of KEPCO for their very thorough preparations for the Symposium and for their dedication to its success. In particular, thanks go to Mr. Michel Vidard of Electricite de France, who has served as Chairman of the Steering Committee, and the many staff including Vice President Mr. Chang-Tong Choi, Vice President Mr. Yong-Taek Park, Mr. Yong-Nam Park, and to Mr. Jang-Hee Hong and his staff particularly Mr. Myung-Sub Noh, Mr. Seong-Gyun Moon, Mr. Young-Chul Lee and Ms. Jee-Hye Kim and many other staff, who have been responsible for the local organization.

Thank you very much.

CLOSING REMARKS

Yong-Nam Park
Korea Electric Power Corporation,
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Good afternoon, distinguished participants, ladies and gentlemen. Now it is time to conclude this Symposium. It is amazing how time passes so fast, like shooting an arrow. I believe that these three days with useful discussions and information exchanges on many issues that were meaningful to all of us, in particular the barriers to nuclear energy deployment were analysed and strategies were discussed to overcome them in various domains such as safety, advanced technologies, economic viability and public acceptance of evolutionary water cooled reactors. We have come to the conclusion that safety and economic viability must be enhanced in order to ensure the sustained deployment of nuclear energy. Also transparency of all nuclear activities must be ensured to secure public acceptance and support. As Director General ElBaradei stated in his opening address, for the future of nuclear power generation, technology development to enhance the safety is very important, and this is our main task. In this context I'm confident that this meeting has served as an impetus to promote further development on new nuclear technologies and to consolidate the international co-operation.

And now I'd like to conclude my closing remarks by thanking all who played an important role in making this symposium successful. First of all I'd like to appreciate the chairpersons, panellists, authors including poster session exhibitors and all participants for their enthusiastic participation during this symposium. Also I would like to thank the OECD/NEA, the Uranium Institute, the Korean Nuclear Society and the Korea Atomic Industrial Forum for their co-operation in organizing this symposium, and ABB, AECL and COGEMA also for their active support.

Lastly, I'd like to express my special thanks to the IAEA, especially Mr. P.E. Juhn, Mr. Juergen Kupitz, Mr. John Cleveland, Mr. Byung-Oke Cho, Mr. Bob Lyon, Ms. Perricos, Ms. Niedermayr and the staff of KEPCO for their dedication and devotion. In particular I'd like to thank Mr. Michel Vidard, Chairman of the Steering Committee, and all his members, and for those of you who have travelled far to attend this symposium I sincerely hope that your stay here in Korea has provided an opportunity to experience the Korean culture and hospitality. I hope you have a safe trip back to your country. Thank you very much.

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