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Status of advanced light water reactor designs 2004

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FOREWORD

The future utilization of nuclear power worldwide depends primarily on the ability of the nuclear community to further improve the economic competitiveness of nuclear power plants while meeting stringent safety requirements. The IAEA's activities in nuclear power technology development include the preparation of status reports on advanced reactor designs to provide all interested IAEA Member States with balanced and objective information on advances in nuclear plant technology.

In the field of light water reactors, the last status report published by the IAEA was "Status of Advanced Light Water Cooled Reactor Designs: 1996" (IAEA-TECDOC-968). Since its publication, quite a lot has happened: some designs have been taken into commercial operation, others have achieved significant steps toward becoming commercial products, including certification from regulatory authorities, some are in a design optimization phase to reduce capital costs, development for other designs began after 1996, and a few designs are no longer pursued by their promoters.

With this general progress in mind, on the advice and with the support of the IAEA Department of Nuclear Energy's Technical Working Group on Advanced Technologies for Light Water Reactors (LWRs), the IAEA has prepared this new status report on advanced LWR designs that updates IAEA-TECDOC-968, presenting the various advanced LWR designs in a balanced way according to a common outline. The objective is to provide Member States, including those considering the initiation of nuclear power programmes, with an overview of development trends and goals, and descriptions of advanced LWR designs for electricity production and for co-generation of electricity and heat, and their status of development.

The IAEA appreciates the advice and support of the members of the IAEA's Technical Working Group on Advanced Technologies for LWRs in the preparation of this status report. Specifically, the IAEA appreciates the support of the following steering group members who guided this activity: E. Patrakka (Teollisuuden Voima Oy, Finland); F. Depisch (Framatome ANP, Germany); N. Fil (Gidropress, Russian Federation); K. Foskolos (Paul Scherrer Institute, Switzerland); and F. Ross and T. Miller (U.S. Department of Energy, United States of America). The IAEA appreciates the information provided by the several organizations involved in development of advanced LWR designs and by groups of potential users who have provided information regarding their requirements. The IAEA officer responsible for this publication was J. Cleveland of the Division of Nuclear Power.

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INTRODUCTION

Nuclear power has proven its viability as an energy source in many countries. Nuclear power technology is mature, and has achieved tremendous progress in the last decades. Worldwide, the installed nuclear capacity at the end of 2002 was 358.7 GW(e), and nuclear plants were operating in 30 Member States[1]. During 2002, nuclear power supplied 16.0% of the world's electricity [1]. A very broad experience of operating nuclear power plants is available, and the total operating experience worldwide, at the end of 2002, was 10,803 reactor-years [2].

Substantial design and development programmes are underway in a number of Member States for further technology improvements and for development of advanced nuclear power plant designs. This development is proceeding for all reactor lines — water cooled reactors, gas cooled reactors, and liquid metal cooled reactors so that nuclear power can play an important and increasing role in global energy supply in the future. Global trends in advanced reactor designs and technology development are periodically summarized in status reports, symposia and seminar proceedings prepared by the IAEA [3–8] to provide all interested IAEA Member States with balanced and objective information on advances in nuclear plant technology.

Worldwide, light water reactors[2] (LWRs), are the dominating type of nuclear plants and, by the end of 2002, had accumulated a total operating experience of 7823 reactor-years. LWRs represent 88.3% of the global nuclear power capacity, and advanced LWR designs building on this experience base are being developed in several countries to help meet future energy needs.

Objective and structure of this report

The objective of this report is to provide Member States, including those considering the initiation of nuclear power programmes, with balanced and objective information on development of advanced LWRs including an overview of development trends and objectives, and descriptions of advanced LWR designs for electricity production and for co-generation of electricity and heat, and their status of development. This report is intended to be a source of reference information for interested organizations and individuals. Among them are decision makers of countries considering implementation of nuclear power programmes. Further, this report is addressed to government officials with technical background and to research institutes of countries with existing nuclear programmes that wish to be informed on the global status in order to plan their nuclear power programmes including both research and development efforts and means for meeting future energy needs. The report is also intended to provide the public with unbiased information on nuclear power.

Chapter 1 provides an overview of the current status and future potential of nuclear power. Chapter 2 provides an overview of trends in advanced LWR design and technology including the means for reducing cost and construction times, improving performance and achieving high levels of safety. Chapter 3 provides a summary of safety objectives for future plants, a summary of activities to prepare user's requirements to guide design efforts for future plants and gives an example of the application of user's requirements by a potential owner of a new nuclear unit. Chapters 4–6 provide descriptions of advanced LWR designs for electricity production and for co-generation of electricity and heat including the nuclear system, the power conversion system, the instrumentation and control system, electrical system, safety concept and summary level technical data according to a common outline. Also presented for the designs are the specific measures that designers have taken, or are taking, to

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[1] The data are available from IAEA's Power Reactor Information System (PRIS). The totals include the nuclear capacity and nuclear electricity generation in Taiwan, China.

[2] Light water reactors are reactors with light water moderator and coolant [i.e. pressurized light water moderated and cooled reactors (PWRs), boiling light water moderated and cooled reactors (BWRs) and water cooled, water moderated power reactors (WWERs)].

simplify the design, to reduce costs, construction schedule and the need for maintenance, to achieve high availability and flexibility of operation, and to improve the ability to maintain the plant, and the measures incorporated into the design for achieving high levels of safety. These descriptions have been provided by the various design organizations involved in development of advanced LWRs. They are intended to present technical descriptions of the various designs and the design organizations' claims regarding the projected performance of their designs.

Goals of nuclear power development

Nuclear power has demonstrated excellent technical and economic performance in many countries, and, like any other progressing technology, it continues to pursue improvements. The accumulated experience is being used to develop advanced nuclear power plant designs. Improved economic competitiveness and enhanced safety are common goals for advanced designs.

Achieving economic competitiveness of future plants

With regard to economic competitiveness, most of the world's electricity markets are moving towards greater competition. Both private sector and state-owned electricity generating organizations carefully examine the costs of their operations, and focus on supply technologies that are low cost and low risk.

It is generally agreed that the largest commercial barrier to the addition of new nuclear power capacity is the high capital cost of nuclear plants relative to other electricity generating alternatives. If nuclear plants are to form part of the future generating mix in competitive electricity markets, capital cost reduction through simplified designs must be an important focus. Reductions in operating, maintenance and fuel costs should also be pursued[3].

Capital costs for nuclear plants generally account for 45–75% of the total nuclear electricity generation costs, compared to 25–60% for coal plants and 15–40% for gas plants. Until recently, nuclear power's advantage in having a small share of its generating costs in fuel costs could offset the disadvantage of its high capital costs. Moreover, in protected markets, investment costs could be recovered over several decades through regulated rates. Now, electricity markets are undergoing major changes. Alternative generating technologies are becoming increasingly efficient, and the capital costs of some alternative technologies per installed kW(e) have fallen significantly. With increased competition in the electric power industry, short term profitability has become a criterion for successful generation along with long term economic viability. With deregulation, owners are not guaranteed cost recovery through regulated rates, and, with privatization, investors seek appropriately rewarded risk, which often translates into seeking small capital investments and high returns, and the minimization of their economic risks, including those arising from political intervention or public opposition.

Design organizations are challenged to develop advanced nuclear power plants with lower capital costs and shorter construction times (e.g. by simplification, standardization, modularization, etc.) and sizes suitable for various grid capacities and owner investment capabilities. This includes large sizes for some markets and small and medium sizes for others. To achieve the largest reductions in capital cost, both proven means and new approaches should be applied. These proven means and new approaches are discussed in more detail in **Section 2.2**, and the various LWR designs under development are presented in **Chapters 4–6**.

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[3]Although the economic competitiveness of fossil fuelled plants may be reduced in the future due to, for example, rising fuel costs and, in some countries, the introduction of taxes on CO_2 emissions, the nuclear power industry should not have a reduced incentive for cost reduction. Importantly, technologies for fossil fuelled plants also progress and one area of current development involves "clean" new plants with carbon capture.

In examinations of economic competitiveness, the external costs of various energy options should also be addressed. In idealized markets all costs associated with a technology would be internalised as part of its economic cost, and decisions based solely on economic costs would automatically properly reflect all social considerations. Nuclear energy is largely ahead of other energy technologies in internalising its external costs. This is discussed further in **Section 1.3** under the sub-topic of nuclear power and sustainable development**.**

Achieving very high safety levels for future plants

Comparative assessments of the health and environmental risks of different electricity generation systems show that nuclear power and renewable energy systems are at the lower end of the risk spectrum [9]. There has been one nuclear power plant accident with significant health impacts — the Chernobyl accident of 26 April, 1986. The Chernobyl plant was based on a very different design approach than LWRs, and, as was stated in Ref. [10], the plant had been designed with an operational mode that could cause the nuclear chain reaction to grow suddenly by a very large factor if it were not stopped immediately. There was no rapid means to stop it under the conditions of the accident. The TMI accident, which occurred on 28 March, 1979, which involved a severe core melt, has shown that the concept of defence-in-depth is an effective approach for protecting the public, although the accident resulted in a major financial loss. As stated in Ref. [11], the accident at Chernobyl demonstrated that the lessons from the Three Mile Island accident had not been acted upon in the USSR: in particular, the importance of systematic evaluation of operating experience; the need to strengthen the on-site technical and management capability, including improved operator training; and the importance of the man-machine interface.

Since the Chernobyl accident, comprehensive exchange of information and operational know-how has become a major factor in nuclear safety improvements worldwide. The activities of the World Association of Nuclear Operators (WANO) 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 safety review and assessment missions, establishment of internationally recognized safety standards and requirements, promotion of safety culture in nuclear installations, and activities within the Convention on Nuclear Safety, are helping to assure a very high level of global nuclear safety. With the exception of the Chernobyl accident, nuclear power plants have operated with a high level of safety over the past half-century – a fact that must be kept in mind in debates about nuclear plant safety. Major efforts have been made to improve nuclear plant safety through the enhancement of nuclear safety culture and the application of advanced technology to improve engineering and design safety features of existing nuclear power plants. The global safety record for nuclear power plants has shown continued improvement, with marked progress in safety-related modernisation of reactors in Central and Eastern Europe.

The safety of future plants will build on experience in achieving the high levels of safety of current plants. The new nuclear power plant designs currently under development incorporate various technical features to meet very stringent safety requirements [12]. Specifically, safety objectives for future plants include reducing the likelihood of accidents as well as mitigating their consequences in the extremely unlikely event that they occur. The objectives include the practical elimination of accident sequences that could lead to large early radioactive release, whereas severe accidents that could imply late containment failure are to be considered in the design process so that their consequences would necessitate only protective measures limited in area and in time [13], [14].

Some new designs rely on well-proven and highly reliable active safety systems to remove decay heat from the primary system and to remove heat from the containment building during accidents. Other new designs incorporate safety systems that rely on passive means using, for example, gravity, natural circulation, and compressed gas as driving forces to transfer heat from the reactor system or the containment to either evaporating water pools or to structures cooled by air convection. Considerable development and testing of passive safety systems has been and is being carried out in several countries. In other designs a coupling of active safety systems and passive safety systems is adopted. For each of the aforementioned approaches, the main requirement is that the proposed safety systems fulfill the necessary functions with appropriate reliability.

In addition, the development of electronics, computers and software, and instrumentation and control (I&C) technology is progressing rapidly, offering opportunities to enhance the safety of nuclear plants. As equipment in current plants is becoming obsolete and is being replaced, experience with new (digital) I&C systems is being obtained through implementation of advanced systems in modernization projects for current plants.

A more detailed discussion of safety goals and requirements for future plants, and of approaches for meeting the requirements, is provided in **Chapter 3**.

Proliferation-resistance of nuclear power

The potential linkage between peaceful use of nuclear energy and the proliferation of nuclear weapons has been a continuing societal concern. To ensure the absence of undeclared nuclear material and activities or diversion of nuclear material for weapons purposes, an international non-proliferation regime has been developed. This regime consists of the following components:

- An international institutional framework for non-proliferation based on the Non-Proliferation Treaty and comprehensive IAEA safeguards agreements and protocols;
- International verification measures (the IAEA Safeguards system plus regional and bilateral agreements) to provide credible assurance of the non-diversion of nuclear material and of the absence of undeclared nuclear material and activities;
- Export controls on nuclear materials, specified facilities, equipment and other materials, including dual-use technologies and materials; and
- National physical protection measures and material accounting and controls measures, as well as IAEA recommendations on physical protection.

It is desirable that IAEA safeguards have a minimal impact on plant operations while ensuring efficient acquisition of safeguards data. With these goals in mind, as designs of nuclear plants and IAEA safeguards techniques have developed, guidelines for plant design measures have been identified by the IAEA [15], which, if taken into account in the plant design phase, would help to ensure efficient acquisition of safeguards data and minimize the impact of the safeguards activities on plant operations. These guidelines[4] are based on IAEA experience in implementing safeguards, as well as on developments in safeguards technology.

Proliferation resistance is defined [16] as *that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States intent on acquiring nuclear weapons or other nuclear explosive devices*. The degree of proliferation resistance results from a combination of, *inter alia*, technical design features, operational modalities, institutional arrangements and safeguards measures. These can be classified as *intrinsic proliferation resistant features* and *extrinsic proliferation resistant features*. Specifically:

1) *Intrinsic proliferation resistant features* are those features that result from technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures; and

2) *Extrinsic proliferation resistance measures* are those measures that result from States' decisions and undertakings related to nuclear energy systems.

[4] These guidelines address, for example, design of the spent fuel pool area to facilitate viewing of the spent fuel assemblies; provisions that facilitate the verification of fuel transfers out of the spent fuel pool; provision of appropriate backup for power supply outages to avoid interruption of power to safeguards equipment; provision of access to appropriate penetrations in the containment building for data transfer lines serving remote safeguards equipment; and other design measures.

Safeguards is an extrinsic measure comprising legal agreements between the party having authority over the nuclear energy system and a verification or control authority, binding obligations on both parties and verification using, inter alia, on-site inspections.

Four general types of intrinsic proliferation resistant features of nuclear energy systems (i.e. nuclear plants and fuel cycle facilities) have been identified in [16] and are, in summary:

1) Technical features that reduce the attractiveness for nuclear weapons programmes of nuclear material during production, use, transport, storage and disposal;

2) Technical features that prevent or inhibit the diversion of nuclear material;

3) Technical features that prevent or inhibit the undeclared production of direct-use material; and

4) Technical features that facilitate verification, including continuity of knowledge. These features include those described in [15].

Approaches for introducing proliferation resistant features into nuclear energy systems include, but are not limited to, the following:

a. **Reliance on the once-through fuel cycle** would reduce fissile material diversion opportunities that might be associated with fuel reprocessing and recycling.
b. **Establishment of energy parks with both nuclear power plants and fuel cycle facilities** would avoid the need to transport fissile material between sites.
c. **Establishment of a closed fuel cycle with reprocessing that returns minor actinides with plutonium to the reactor for consumption**, could avoid the separation of minor actinides from fissile material so that the material is not weapons useable.
d. **Operating reactors with long operating cycles (e.g., several years) without refuelling or fuel shuffling** could assure that fissile material in the core is not accessible as long as the reactor vessel is not opened. Some new design concepts include the measure that the reactor be returned to the supplier country for refuelling.
e. **Incorporating features to increase the difficulty of extracting fissile material from fresh or spent fuel**.
f. **Incorporating features that greatly reduce the fraction of plutonium in spent fuel** would require that a very large volume of spent fuel would need to be processed to extract sufficient plutonium for a nuclear weapon.
g. **Reducing the fuel stored at a site** would reduce the amount of material that could potentially be diverted from that site.
h. **Reducing the fissile material produced in the reactor** could reduce the weapons-useable material in spent fuel.

It is important to note that some approaches are mutually incompatible in the sense that one approach may not allow, or may be detrimental to, another approach.

Also, there are drawbacks associated with some of the above approaches. For example, the once-through fuel cycle does not allow nuclear energy to become a long term sustainable source of energy. Operating reactors with long fuel cycles of several years requires higher fuel enrichment and the parasitic absorption of neutrons by fission products reduces the fuel utilization efficiency. Features that greatly increase the difficulty of extracting fissile material from spent fuel can create a cost penalty on fuel reprocessing.

The technical and analytical evaluations that were conducted by forty IAEA Member States and four international organizations within the International Nuclear Fuel Cycle Evaluation (INFCE) [17] also provide useful information with regard to proliferation concerns. It was recognized that nuclear energy for peaceful purposes is not the only or perhaps the preferred route for misuse for the purposes of constructing a nuclear weapon. The INFCE evaluations considered whether further technical, institutional or improved safeguards measures might be introduced to discourage such misuse. INFCE identified ways and means of strengthening assurances of nuclear energy supply while at the same time minimizing the risk of the proliferation of nuclear weapons. These would involve international cooperation and include technical, legal and institutional measures as well as continuing developments in the field of safeguards. As stated in Ref. [17], "the findings of INFCE strengthened the view that:

- nuclear energy is expected to increase its role in meeting the world's energy needs and can and should be widely available to that end;
- effective measures can and should be taken to met the specific needs of developing countries in the peaceful uses of nuclear energy; and
- effective measures can and should be taken to minimize the danger of the proliferation of nuclear weapons without jeopardizing energy supplies or the development of nuclear energy for peaceful purposes".

Overview of global development of advanced nuclear plant designs

New generations of nuclear power plants are being developed, building upon the background of nuclear power's success and applying lessons learned from the experience of operating plants. Various organizations are involved in this development, including governments, industries, utilities, universities, national laboratories, and research institutes.

The advanced designs generally incorporate improvements of safety features, including, among others, features that increase the prevention of accidents including features that will allow operators more time to assess the situation before acting, and features that will provide even more protection against any possible releases of radioactivity to the environment. Great attention is also paid to making new plants simpler to operate, inspect, maintain and repair, thus increasing their overall cost efficiency.

Advanced designs comprise two basic categories. The first category is called evolutionary designs and encompasses direct descendants from predecessors (existing plant designs) that feature improvements and modifications based on feedback of experience and adoption of new technological achievements, and possibly also introduction of some innovative features, e.g., by incorporating passive safety systems. Evolutionary designs are characterized by requiring at most engineering and confirmatory testing prior to commercial deployment. The second category consists of designs that deviate more significantly from existing designs, and that consequently need substantially more testing and verification, probably including also construction of a demonstration plant and/or prototype plant, prior to large scale commercial deployment. These are generally called innovative designs. Often a step increase in development cost arises from the need to build a prototype reactor or a demonstration plant as part of the development programme (see Figure 1).

The IAEA differentiates nuclear plants of various power levels by classifying them as:

- Large-size designs: 700 MW(e) and larger
- Medium-size designs: 300 –700 MW(e)
- Small-size designs: below 300 MW(e).

In the near term most new nuclear plants will likely be evolutionary designs building on today's successful proven systems while incorporating technology advances and often pursuing economies of scale. In the longer term, development and demonstration of new, innovative designs, including their promised short construction and startup times and low capital costs, could help to promote a new era of nuclear power.

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. The latter 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 minimize technological risks. The development of an evolutionary design requires at most engineering and confirmatory testing.

Innovative design

An innovative design is an advanced design that 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.

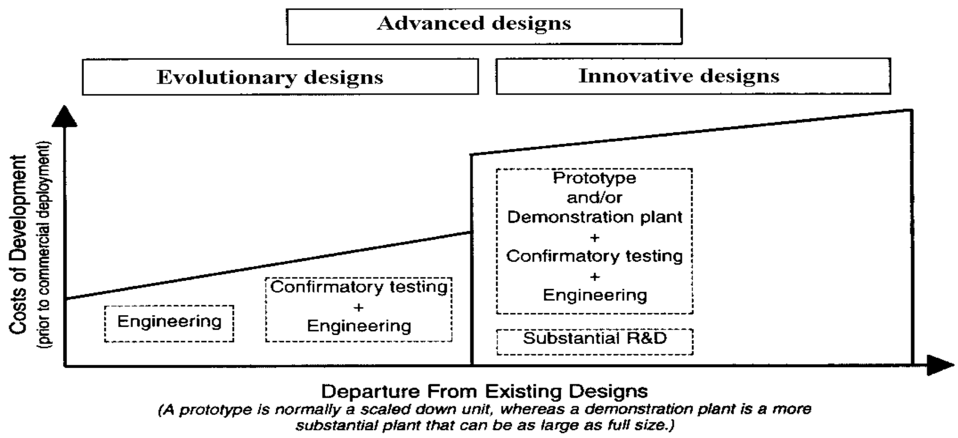

FIG. 1. Efforts and development costs for advanced designs versus departure from existing designs (Terms are excerpted from Ref.[18]).

Several innovative designs are in the small-to-medium size range and would be constructed with factory built structures and components, including complete modular units for fast on-site installation. Small-to-medium size reactors have the potential to capture economies of series production instead of economies of scale, if several units are constructed. Such smaller and easier to finance systems would be particularly attractive for countries with small electricity grids or remote locations. They could also be used for district heating, sea water desalination, hydrogen production, and other non-electric applications.

Advanced nuclear plant designs presently under development comprise the following basic reactor types:

- water cooled reactors, utilizing water as coolant and moderator. These are comprised of light water reactors (LWRs), which use light water as both the coolant and the moderator, and heavy water reactors (HWRs), which use heavy water as moderator and either light or heavy water as coolant;
- gas cooled reactors, using helium as coolant and graphite as moderator; and
- fast reactors, using liquid metal (e.g. sodium) or gas (helium) as coolant.

While this TECDOC addresses the status of advanced LWR designs, a summary overview of global development activities for all advanced reactor types is given below to place the development of advanced LWRs into the context of the global development of nuclear energy.

Light water reactors

80.5% of the total number of nuclear units in operation worldwide are LWRs. Advanced LWRs are being developed over a large range of power levels.

Some examples of large evolutionary LWR designs are: the ABWR and the ABWR-II of General Electric (GE), USA, and Hitachi and Toshiba, Japan; the APWR of Westinghouse, USA and Mitsubishi, Japan and the APWR$^+$ of Mitsubishi; the BWR 90+ of Westinghouse Atom, Sweden; the EPR and the SWR-1000 of Framatome ANP, France and Germany; the ESBWR of GE, USA; the AP-1000 of Westinghouse, USA; the WWER-1000 and the WWER-1500 of Atomenergoproject and Gidropress, Russia; the KSNP$^+$ and the APR-1400 of Korea Hydro and Nuclear Power and the Korean Nuclear Industry, the Republic of Korea; and the CNP-1000 of China National Nuclear Corporation, China.

Among the small and medium-size LWRs, typical designs are: the AP-600 and the integral IRIS design of Westinghouse USA; the WWER-640 of Atomenergoproject and Gidropress, the PAES-600 of OKBM and the VK-300 of RDIPE Russia; and the HSBWR and HABWR design concepts of Hitachi, Japan; and the NP-300 of Technicatome, France.

Two ABWRs are operating at TEPCO's Kashiwazaki-Kariwa site, and deployment programmes are underway for 10 more ABWR units in Japan. Two ABWR units are under construction in Taiwan, China. The EPR, the SWR-1000, the WWER-1000 and the ABWR have been offered commercially for the 5th nuclear plant in Finland. In October 2003, Teollisuuden Voima Oy (TVO) announced that it has selected the Olkiluoto site for the new nuclear unit and that the EPR is the preferred alternative. Also, Electricite de France and the French Government are considering construction of the EPR in France[5]. In the Rep. of Korea, the first units of KSNP$^+$ are planned for Shin-Kori Units 1 and 2 with start of construction in 2004 and 2005 respectively. The plan for the first of two APR-1400 units at

[5] In November 2003, the French government endorsed a new nuclear programme with the publication of a White Paper on energy policy that calls for keeping the nuclear option open by building a demonstration unit based on the EPR. Following a period for public comment, the final version of the White Paper is expected to be adopted in the Council of Ministers and submitted to the French parliament in early 2004 for debate.

Shin-Kori is to start construction in June 2005 with commissioning in 2010. Westinghouse, in March 2002, submitted an application to the U.S. NRC for Final Design Approval and Design Certification of the AP-1000. The Final Design Approval is expected in 2004 and Design Certification is expected in 2004/2005. The IRIS design is in the first phase of pre-application licensing in which the NRC will provide feedback on necessary testing and an assessment of the risk-informed regulation approach. The plan is to submit an IRIS Design Certification application in 2005, with the objective of obtaining design certification in 2008/2009. In mid-2002 the ESBWR design and technology base were submitted to the U.S. NRC with the objective of obtaining closure of all technology issues in 2003, as a first step toward obtaining Design Certification. For Framatome's SWR-1000, in the U.S., the pre-application phase for Design Certification by the U.S. NRC has been started in 2002.

Designers of innovative designs, for example, some of the designs with an integral primary system, such as SMART of the Republic of Korea and CAREM of Argentina, are planning verification and prototype or demonstration plants prior to their commercial deployment. Examples of innovative LWR designs for high conversion of fertile isotopes to fissile isotopes are the RMWR of JAERI and the RBWR of Hitachi, Japan.

A prototype or a demonstration plant will most likely be required for thermodynamically supercritical water cooled systems, which have been selected for development by the Generation-IV International Forum (see the summary discussion of the Generation-IV International Forum (GIF) later in this Introduction). The SCPR concept being developed by Toshiba, Hitachi and the University of Tokyo is an example thermodynamically supercritical LWR.

Heavy water reactors

HWRs account for about 8% of the nuclear power reactors that are currently operating. Two types of commercial pressurized heavy water cooled reactors have been developed, the pressure tube and the pressure vessel versions. HWRs with power ratings from a few hundred MW(e) up to approximately 900 MW(e) are available. The heavy water moderation yields a good neutron economy and has made it possible to utilize natural uranium as fuel. Both the pressure tube and pressure vessel designs use on-load refuelling.

In Canada, the approach taken by AECL in development of next generation CANDU plants (the ACR-700) is to essentially retain the present evolutionary CANDU reactor characteristics and power levels (e.g. the CANDU-6 and CANDU-9 with net electric power levels around 650 MW(e) and 900 MW(e) respectively) and to improve economics through plant optimization and simplification. The ACR-700 design uses slightly enriched uranium and light water coolant. It is currently undergoing a pre-application licensing review by the US Nuclear Regulatory Commission. Following that review, AECL intends to seek a Design Certification in 2005. The ACR-700 is simultaneously undergoing a licensing review in Canada.

Also, in Canada in the framework of GIF, AECL is developing an innovative design, the CANDU-X, which would use supercritical light water coolant to achieve high thermodynamic efficiency.

In India, a continuing process of evolution of HWR design has been carried out since the Rajasthan 1 and 2 projects. In 2000 construction began on two 540 MW(e) units at Tarapur which incorporate feedback from the indigenously designed 220 MW(e) units[6].

India is also developing the Advanced Heavy Water Reactor (AHWR), a heavy water moderated, boiling light water cooled, vertical pressure tube type reactor, optimized for utilization of thorium for power generation, with passive safety systems.

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[6] The most recent plants in this series, the 220 MW(e) Kaiga-1 and the Rajasthan-3 and –4 units, were connected to the grid in the year 2000.

Reference [3] provides a detailed discussion of the status and projected development of HWRs.

Gas cooled reactors

Gas cooled reactors have been in operation for many years. In the United Kingdom (UK), the nuclear electricity is mostly generated by CO_2 cooled Magnox and advanced gas cooled reactors (AGRs). Development of high temperature reactors (HTGRs) with helium as coolant, and graphite as moderator, has also been going on for a long period of time. Prototype and demonstration plants with the Rankine steam cycle for electric power generation have been built and operated.

The inert He coolant and the coated fuel particle design enable HTGRs to operate at temperatures considerably above those in water cooled reactors. Development is also conducted for high temperature heat applications. Currently two helium cooled test reactors are in operation. The High Temperature Engineering Test Reactor (HTTR) at the Japan Atomic Energy Research Institute (JAERI) in Japan and the HTR-10 at the Institute of Nuclear Energy Technology (INET) in China.

Presently, a considerable effort is devoted to the gas-turbine direct cycle, pebble bed small-size modular HTR (PBMR) that promises high thermal efficiency and low power generation cost. Eskom, South Africa's Industrial Development Corporation, and BNFL (United Kingdom) are jointly developing such a system. Also, the Ministry of the Russian Federation for Atomic Energy, the Experimental Design Bureau for Machine Building (OKBM), General Atomics, Framatome and Fuji Electric are jointly developing a small gas turbine modular helium reactor (GT-MHR) for electricity production and the consumption of weapons grade plutonium.

A helium cooled Very High Temperature Reactor (VHTR) with a focus on hydrogen production is being developed within the framework of GIF.

Fast reactors

Liquid metal cooled fast reactors (LMFRs) have been under development for many years in a number of countries, primarily as breeders. The successful design, construction and operation of several sodium-cooled reactor plants, such as the small size Prototype Fast Reactor in the United Kingdom, the prototype Phénix fast reactor in France, the BN-350 in Kazakstan (part of its thermal energy was used for sea water desalination), both the demonstration BN-600 in Russia, and the Monju in Japan, as well as the commercial size Superphénix in France, have provided an extensive experience base of more than 200 reactor-years for further improvements. In addition, this is a considerable base of experience with lead-bismuth (eutectic) cooled propulsion (submarine) reactors built and operated in the former USSR.

Fast reactors use fast neutrons for sustaining the fission process, and they can actually produce fuel, as well as consuming it. Plutonium breeding allows fast reactors to extract sixty-to-seventy times as much energy from uranium as thermal reactors do. Their capability to produce more fissile material than they consume may become indispensable in the longer term if the deployment of nuclear power is increased substantially. Fast reactors may also contribute to reducing plutonium stockpiles, and to the reduction of the required isolation time for high level radioactive waste by utilizing transuranic radioisotopes and transmuting some of the most cumbersome long lived fission products.

Examples of current LMFR activities include: the construction in China of the small size Chinese Experimental Fast Reactor (CEFR) with first criticality scheduled for 2006; the development of the small-size KALIMER design in the Republic of Korea; the successful operation of the Indian Fast Breeder Test Reactor (FBTR) and its utilization for fast reactor R&D, especially fuel irradiation and materials research; the development of the medium size Prototype FBR (PFBR) in India for which construction has started in 2003; efforts in Japan aimed at restarting MONJU, and the Japan Nuclear Cycle Development Institute's "Feasibility Study on a Commercialised Fast Reactor Cycle System"; efforts in Russia to complete the BN-800 reactor at Beloyarsk by 2010, and design studies of advanced fast reactors (sodium cooled, lead cooled, and lead-bismuth eutectic cooled) having improved economics and enhanced safety.

In France, the Phénix plant has restarted in 2003 with the main mission of conducting experiments on long lived radioactive nuclide incineration and transmutation.

Development activities for a gas (helium) cooled fast reactor (GFR) with an integrated fuel cycle with full actinide recycle and for lead alloy and sodium cooled systems are being conducted within GIF.

Co-operative international research is underway in several countries on fast neutron spectrum hybrid systems (e.g., accelerator driven systems (ADS)). The potential advantages of ADS systems are low waste production, high transmutation capability, enhanced safety characteristics and better long term utilization of resources (e.g., with thorium fuels). ADS research activities include development of the HYPER concept by the Republic of Korea; design studies and research on basic physical processes in Russia and in eight countries in the European Union, and in the Advanced Accelerator Applications programme of the U.S.A. (recently merged with the Advanced Fuel Cycles initiative).

International initiatives for innovative plants

Many countries believe that nuclear energy must remain or become an integral part of their energy mix to meet energy supply needs. To help achieve this goal, there are two major international efforts, the Generation IV International Forum and the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO).

Concerns over energy resource availability, climate change, air quality, and energy security suggest an important role for nuclear power in future energy supplies. While the current Generation II (commercial power reactors) and Generation III ([currently available] advanced LWRs) nuclear power plant designs provide an economically, technically, and publicly acceptable electricity supply in many markets, further advances in nuclear energy system design can broaden the opportunities for the use of nuclear energy.

To explore these opportunities, the U.S. Department of Energy's Office of Nuclear Energy, Science and Technology has engaged governments, industry, and the research community worldwide in a wide-ranging discussion on the development of next-generation nuclear energy systems known as "Generation IV". This has resulted in the formation of the Generation-IV International Forum (GIF), a group whose member countries are interested in jointly defining the future of nuclear energy research and development. Members are Argentina, Brazil, Canada, Euratom, France, Japan, the Republic of Korea, South Africa, Switzerland, the United Kingdom and the United States. The IAEA and the OECD/NEA have permanent observer status in the GIF Policy Group, which governs the project's overall framework and policies. In short, "Generation IV" refers to the development and demonstration of one or more Generation IV nuclear energy systems that offer advantages in the areas of economics, safety and reliability, sustainability, and could be deployed commercially by 2030.

As stated in [19] the purpose of the GIF and the Vision for Generation IV is "The development of concepts for one or more Generation IV nuclear energy systems that can be licensed, constructed, and operated in a manner that will provide a competitively priced and reliable supply of energy to the country where such systems are deployed, while satisfactorily addressing nuclear safety, waste, proliferation and public perception concerns." Following evaluations of many concepts, six systems have been selected by the GIF Policy Group for future bilateral and multilateral cooperation, and a Technology Roadmap has been prepared to guide the research and development [20]. The six selected systems are:

- Gas cooled fast reactor systems
- Lead alloy liquid metal cooled reactor systems
- Molten salt reactor systems
- Sodium liquid metal cooled reactor systems
- Supercritical water cooled reactor systems
- Very high temperature gas reactor systems

Appendix 1 summarizes information from Ref. [21] to provide brief descriptions of the six Generation-IV systems. An example super-critical water cooled reactor, the SCPR, is described in **Chapter 4** of this TECDOC.

The IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) is based on an IAEA General Conference resolution in September 2000 inviting all interested Member States, both technology suppliers and technology users, to consider jointly international and national actions required to achieve desired innovations in nuclear reactors and fuel cycles. Additional endorsement came in a UN General Assembly resolution in December 2001 that emphasized "the unique role that the Agency can play in developing user requirements and in addressing safeguards, safety and environmental questions for innovative reactors and their fuel cycles" and stressed "the need for international collaboration in the development of innovative nuclear technology".

The main objectives of INPRO are (a) to help to ensure that nuclear energy is available to contribute to fulfilling energy needs in the 21st century in a sustainable manner, and (b) to bring together technology holders and technology users, to consider jointly the international and national actions required to achieve desired innovations in nuclear reactors and fuel cycles. As of December 2003, members of INPRO include Argentina, Brazil, Bulgaria, Canada, China, Germany, India, Indonesia, the Republic of Korea, the Russian Federation, Spain, Switzerland, the Netherlands, Turkey, and the European Commission. In its first Phase, INPRO has prepared guidelines for the evaluation of innovative nuclear reactors and fuel cycles addressing economics, sustainability and environment, safety of nuclear installations, waste management, proliferation resistance as well as cross-cutting issues. [16].

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CHAPTER 1. CURRENT STATUS AND FUTURE POTENTIAL OF NUCLEAR POWER

1.1. Current status of nuclear power utilization

In the past 50 years, nuclear power has grown from a new scientific development to become a major part of the energy mix in 30 Member States. In 2002, 441 power reactors worldwide produced 2574.2 billion kWh of electricity [1], slightly up on 2001 output [2]. Sixteen countries relied on nuclear power for 25 percent or more of their electricity. At the end of 2002, 32 nuclear power plants were under construction: 10 in Eastern Europe; 12 in the Far East; 9 in the Middle East and South Asia; and 1 in Latin America. Figure 2 shows the percent of electricity produced by nuclear power in each country, and Figure 3 shows the number of reactors operating in each country.

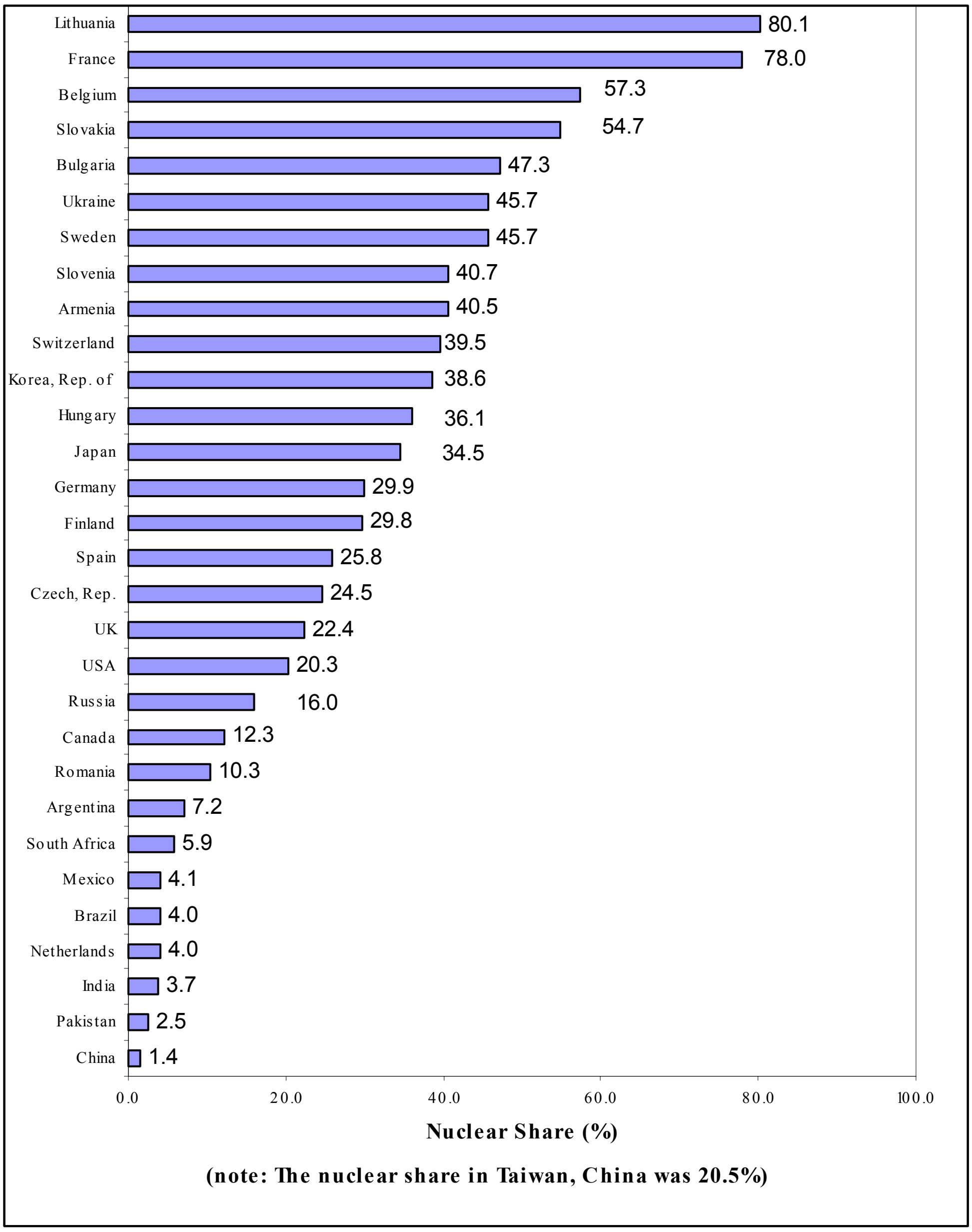
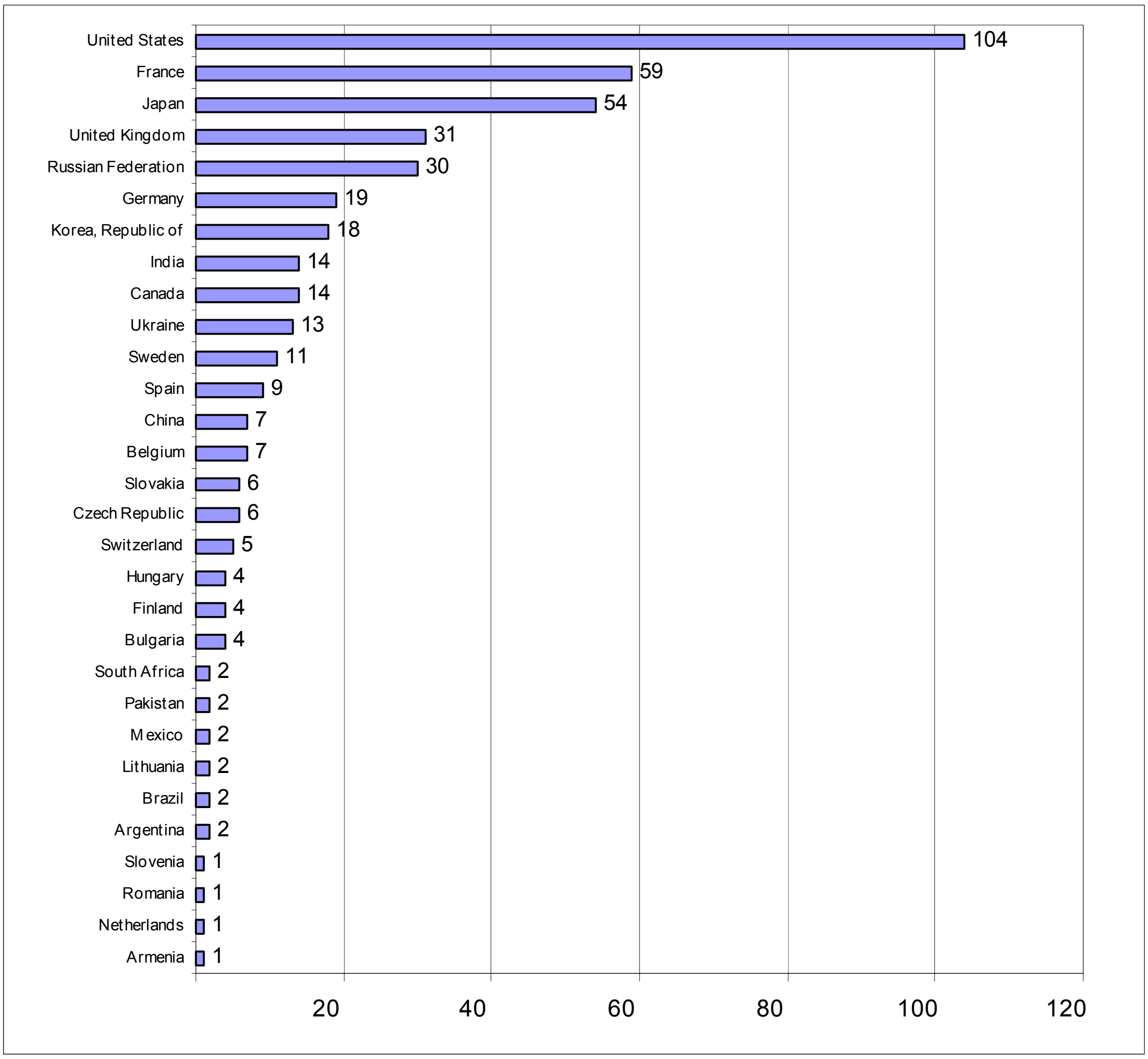


FIG. 2. Percent of electricity generated with nuclear power in 2002 (data from Ref: [1]).



(Note: Six reactors are in operation in Taiwan, China)

FIG. 3. Number of nuclear power plants in operation.

1.2. Benefits of nuclear power

Choices of future energy options are generally made to satisfy the following needs:

- affordably priced and convenient energy;
- adequate energy for growth and development;
- energy security; and
- environmentally benign and low risk sources.

Nuclear power has a proven record of helping many countries meet these needs. Many existing plants are achieving high availability, and are producing power at low and competitive production costs (fuel + O&M). Many plants are nearly or fully amortized, resulting in increased profitability. Power up-ratings and lifetime extension are often economically attractive options. Furthermore, except for nuclear power and hydro-power (which has limited growth potential), there are currently no other economically viable, minimal-greenhouse gas-emission options for base load electricity generation.

In the context of economic competitiveness, the external costs of various energy options should be addressed. Nuclear energy is largely ahead of other energy technologies in internalising its external costs. The costs of waste disposal, decommissioning and meeting safety requirements are in most countries already included in the price of nuclear electricity. Progress towards a more level playing field where external costs of other energy technologies are more consistently internalised as part of their economic costs would thus result in more balanced assessments of energy options. As indicated by the results of the ExternE studies in Europe [3], external costs for fossil-fired plants operated to current standards are well above external costs of NPPs, also operated to current standards.

Nuclear energy's advantages with regard to greenhouse gas (GHG) emissions and energy supply security[6] both received increased attention in, among other policy deliberations, the US energy policy published in May 2001 [4], the European Commission's Green Paper in November 2000 [5] on energy supply security, and the European Parliament resolution in November 2001 [6] reinforcing the role of nuclear energy in avoiding GHG emissions and increasing supply security.

These considerations would suggest that nuclear power's share of global electricity generation would grow as a part of many national energy strategies for sustainable development, together with increased use of renewable sources and improved efficiency throughout the energy system. However, IAEA statistics show a mixed picture. Although nuclear power plants are operating very efficiently, reliably and safely in more than 30 countries, some of these countries have not had new construction starts for many years, and several have policies to phase out their nuclear power. There are currently no new nuclear power projects in North America, although in several countries in Asia and in parts of Eastern Europe nuclear power continues to grow. In some countries, public concern over nuclear safety and nuclear waste disposal are critical factors inhibiting decisions to construct new plants and, in some cases, threatening the continued operation of existing safe and efficient plants. And in some countries which are deregulating their energy markets, the high capital costs of nuclear plants and the low price of natural gas in recent years have tended to result in investment in low capital cost systems such as combined cycle gas power plants, which have short construction periods.

The slowdown in new plant construction in the past years can be brought into focus by comparing it with the decade of the 1970s, during which construction was started on 251 units, an average of 25 units per year. In contrast, in 1998, there were only 3 construction starts [China (2), Japan (1)]; in 1999 there were 5 [China (1), Rep. of Korea (2) and Taiwan, China (2)]; in 2000, there were 5 [China (1), India (2), Japan (2)]; in 2001, there was only 1 [Japan]; and in 2002, there were 6 [India (5) and the Democratic Peoples Republic of Korea (1)].

To assure that the long term potential of nuclear energy can be fully exploited, the nuclear community must not only meet the economic challenge. It must also meet the challenges of achieving acceptance of nuclear power in international discussions on climate change as a technology compatible with sustainable energy development, and achieving improved public understanding in all areas. Clearly nuclear power can put the world's large uranium resource to productive use, can reduce harmful emissions associated with burning fossil fuels, and can expand electricity supplies. To be a truly sustainable energy supply, in addition to being economically competitive, nuclear power must implement a long term solution to disposal of high level radioactive waste, continue to achieve the highest level of safety for nuclear plants and for fuel cycle facilities, and assure strong vigilance in security and safeguards of nuclear material.

An important event in 2002 concerning nuclear power and the environment was the World Summit on Sustainable Development, which was held in August, in Johannesburg, South Africa. It produced the Plan of Implementation and the Johannesburg Declaration. Both emphasize and recognize the importance of energy, for the first time at the World Summit, as an essential prerequisite for poverty eradication and socio-economic development. This echoes the decisions in 2001 of the Ninth Session of the Commission on Sustainable Development, called CSD-9, and contrasts notably with the absence of any energy chapter in *Agenda 21*.

The word "nuclear" appears in neither the Plan of Implementation nor the Johannesburg Declaration. However, in the section dealing directly with energy, the Plan of Implementation begins with an explicit call to implement the recommendations and conclusions of CSD-9. With respect to nuclear power, CSD-9's broad conclusions were that countries agree to disagree on the role of nuclear power in sustainable development, and that "the choice of nuclear energy rests with countries". Therefore, what was important for nuclear in Johannesburg was that the essential role of energy for sustainable development was recognized and emphasized, and that the Plan of Implementation reinforced CSD-9's nuclear conclusions — agreeing to disagree and that the choice of nuclear is up to the countries.

Careful evaluations of potential additions of nuclear power plants address the following topics:

- the demand and the demand forecast for base load power;
- certainty of the security of electricity supply;
- generating cost comparisons with other base load supply options, considering cost sensitivities; and
- the national target for reduction in greenhouse gas emissions.

The evaluations conducted in Finland regarding a possible 5th nuclear unit carefully examined these topics. Extensive studies performed with regard to building a new nuclear plant are documented in the paper by Teollisuuden Voima Oy (TVO), published in Ref. [7]. In Finland, the Nuclear Energy Act requires that a company considering a nuclear plant project must apply for a decision in principle from the Government beforehand. This application was submitted on 15 November 2000 by TVO to the Council of State. The actual investment decision can be made after a positive decision in principle has been received from the Council of State and the Parliament. On 24 May 2002, the Finnish Parliament decided to ratify the favorable decision-in-principle made by the Government in January 2002. This decision will be in force for 5 years from the date of its ratification, and within that period, TVO may submit an application for a construction permit, in accordance with the Nuclear Energy Act. In 2002 TVO invited bids for the new unit from NPP suppliers [8], and in October 3003, TVO announced that it has selected the Olkiluoto site and that the EPR is the preferred alternative.

Thus, with construction projects underway in several countries, with plans for a fifth nuclear power plant in Finland, with the possibility of new nuclear units in France and the USA, and with further planned additions mainly in the Far East, nuclear power is proceeding.

In the longer term, recycling the fissile content of spent fuel and breeding additional fissile material from the world's resources of U^{238} and Th^{232} can extend the energy resource available from uranium for centuries. This long term energy strategy will be supported by fast breeder reactors. Also, LWRs with high conversion ratios are being developed with goals of assuring a long term energy supply as well as reducing spent fuel accumulation. It is noted that there is a fast spectrum version of the Generation-IV super-critical water cooled reactor system in order to address long term fuel resource availability goals with water cooled reactor systems. [See also the descriptions of the Reduced Moderation Light Water Reactor (RMWR) and the Resource-Renewable BWR (RBWR) in **Chapter 4**.]

1.3. Nuclear energy projections into the future

Clearly, the contribution of nuclear energy to near and medium term energy supplies will depend on several key issues. The degree of global commitment to sustainable energy development strategies and recognition of the role of nuclear energy in sustainable development strategies will impact its future use. Technological maturity, economic competitiveness, financing arrangements and public acceptance are key factors influencing decisions to build new plants. Public perception of energy options and related environmental issues as well as public information and education will also play a key role in the introduction of evolutionary designs. Continued vigilance in nuclear power plant operation, continued enhancement of safety culture, and preserving intellectual capabilities in nuclear power technology are highly important in preserving the potential of nuclear power to contribute to future energy strategies.

Worldwide, energy use is projected to increase substantially throughout the 21st century. The driving forces are population growth and economic development, particularly in developing countries. The IAEA Nuclear Technology Review [9] addressed the world's rising energy and electricity demand by examining comprehensive projections published by the United Nations Intergovernmental Panel on Climate Change (IPCC) Special Report on Emissions Scenarios (SRES) in 2000. In the 40 reference scenarios in SRES, global primary energy use grows between 1.7 and 3.7-fold by 2050, with a median increase by a factor of 2.5. Electricity demand grows even faster because economic growth consistently prompts a shift towards electricity. By 2050, the IPCC's projected electricity growth from the 2000 level is by a factor of between two and eight. The median increase is by a factor of 4.7.

By design none of the 40 SRES scenarios includes policies to mitigate climate change. However, the IPCC's subsequent Third Assessment Report (TAR) [10], contains results for 76 "post-SRES stabilization scenarios" that incorporate policies to limit carbon emissions resulting in reduced energy growth compared to the SRES scenarios. The vertical bars in Figure 4 show the resulting projections for nuclear capacity from SRES cited above and 19 selected TAR scenarios that have been analysed in Ref. [11]. For the SRES scenarios in 2050 the nuclear capacity projections range between 350 GW(e) up to more than 5,000 GW(e), (with a median of more than 1,500 GW(e)). This would require adding 50–150 GW(e) per year from 2020–2050. For the 19 TAR scenarios, projections for 2050 range from 2300 GW(e) of nuclear capacity up to more than 7000 GW(e) (with a median of more than 4400 GW(e)). This would require additions of 50–205 GW(e) per year from 2020–2050, above and beyond new NPPs required to replace old NPPs retired during that period.

Figure 4 also shows two distinctly lower intermediate term projections for nuclear power. First are recent projections of the IAEA [1], shown by the yellow triangle, based on annual IAEA reviews of specific projects, national plans and national projections.

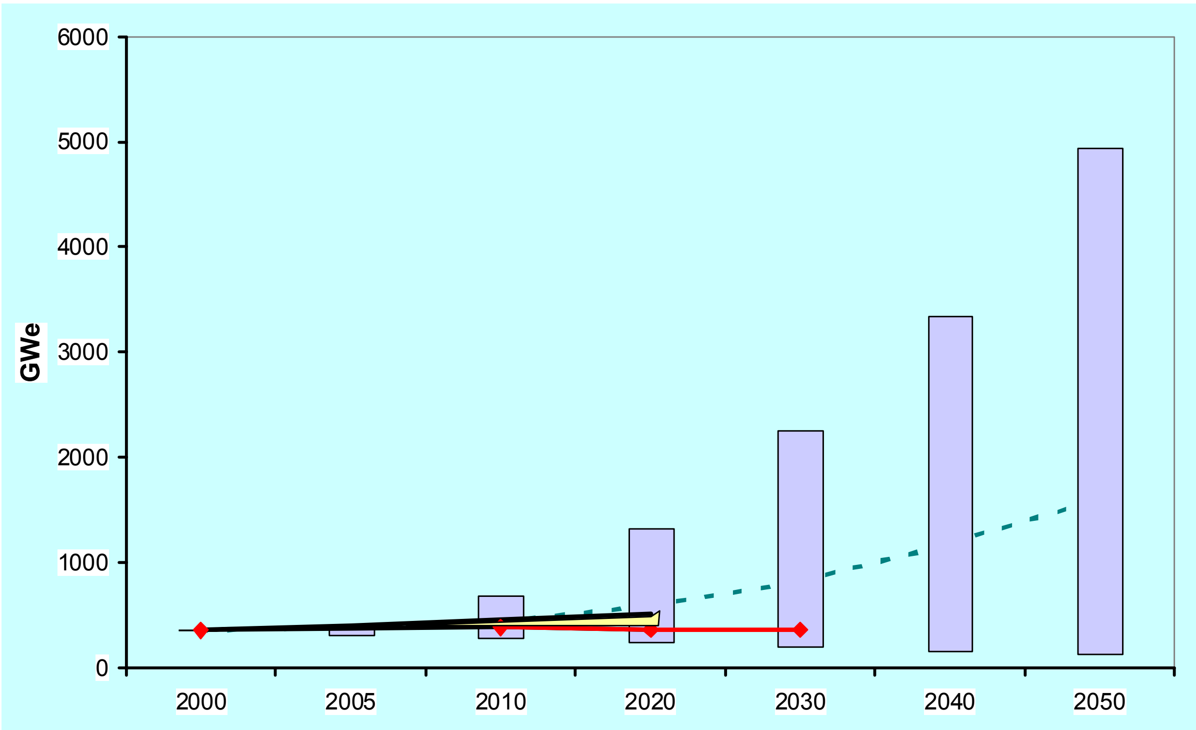


FIG. 4. Nuclear capacity projections from three sets of scenarios: The vertical bars show the range projected in SRES, with the dotted line indicating the SRES medians. The yellow triangle shows the range between the IAEA high and low projections through 2020 from Ref. [1] and the red line shows the IEA projections.

The IAEA high projection estimates a 40% increase in nuclear capacity by 2020, compared with nuclear capacity in 2002, and is only slightly lower than the SRES median. The IAEA low projection estimates an increase in nuclear capacity by 2020 of 18% compared to the capacity in 2002. The last projection in Fig. 4 (red line) is the International Energy Agency reference case [12]. The IEA projects that the global nuclear electricity generation capacity in 2030 will be 356 MW(e), i.e. essentially the same nuclear capacity as in 2002.

The IAEA's projections over the next 25–30 years for electricity production show that it is not expected to keep pace with global growth of electricity demand. Nuclear power produced 16 percent of the world's electricity in 2002, and the IAEA estimates that nuclear power will provide about 15 – 16 % of the world's electricity in 2010. However, IAEA estimates that nuclear power will produce only 13 to 15 % of the world's electricity by 2020, and 11 to 12 % by 2030 [1] as countries invest in other energy options. Although the IAEA estimates that the percentage of the world's electricity produced by nuclear power will decrease, it estimates that the actual amount of nuclear generated electricity will increase. Compared to the 2574.2 billion kWh produced in 2002, the IAEA estimates that nuclear power will produce between 2830 and 2987 billion kWh annually by the year 2010, between 3085 and 3756 billion kWh annually by the year 2020, and between 2881 and 4369 billion kWh annually by the year 2030 [1].

The "projection gap" between the intermediate term scenarios of the IAEA and IEA and the long term scenarios in SRES and TAR is largely due to differing assumptions about political constraints, cost improvements and innovation. The IEA and low IAEA projections assume hostile or indifferent political environments, no innovation and little or no progress on new NPP costs. The long term SRES and TAR scenarios, in contrast, assume, first, that nuclear technologies, like other technologies, are not static and, second, that in the long view investments are made ultimately on the basis of economics. In these scenarios the nuclear industry makes continuing cost reductions, introduces innovations and is able to sell NPPs based exclusively on their cost and performance in a politically neutral market. A number of the SRES and TAR scenarios also assume that an increasing share of nuclear energy is used for innovative non-electric applications, including hydrogen production for both the transport sector and stationary uses.

Closing upwards the projection gap in Figure 4 requires success on several fronts. The nuclear industry must continually reduce costs through evolutionary and innovative improvements, and appropriate financing arrangements must be available for these capital-intensive projects. Progress must also be made on political and public acceptance issues. These factors are discussed below in more detail, and Chapters 4, 5 and 6 present the progress on new designs. The needed success will not happen by itself. Continuing innovative R&D on the part of industry and governments is required, as are financing approaches tailored to electricity market conditions that range from competitive, privatised markets to conditions in developing countries wishing to begin nuclear power programmes and where large amounts of capital are difficult to raise. Progress in political and public acceptance will require continuing public and political discussion of the pros and cons of all energy options, and the varying national priorities of different countries.

Technological readiness

With regard to technological readiness, the large base of experience with the current nuclear plants, and results from research and development programmes, guide development of new designs on the basis of User Requirements Documents (URDs) such as the Electric Power Research Institute URD [13] and the European Utility Requirements (EUR) [14]. Common goals are high availability, good operating features, competitive economics and compliance with internationally recognized safety objectives. The documents require improved performance for future plants: for example, they specify plant availability factors of 87% and above. There is a general drive for simplification, large margins to limit system challenges, long grace periods for response to emergency situations, and use of advanced information technology in the man-machine interface systems. New designs incorporate features to meet stringent safety objectives for improving severe accident prevention and mitigation.

Several of these designs have reached a high degree of maturity. Some designs have been certified by nuclear regulatory authorities. Some are entering a design optimization phase to reduce capital cost. In certain cases design optimization is leading 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. Many of the new design features have been tested to demonstrate technological readiness.

Technological progress is continuing, and there is no doubt that evolutionary reactors will offer improved performance. This can be clearly seen by the steady improvements in performance achieved by current plants. The average energy availability factor for nuclear plants has increased from approximately 70 percent in 1983 to 83.7 percent[7] in 2002, with some utilities achieving significantly higher values. This is being achieved through integrated programmes incorporating personnel training, quality assurance, improved maintenance planning, as well as technological advances in plant 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 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 [15], [16], [17], are examples of international co-operation to improve plant performance.

Improved performance at current plants is supported, for example, by better man-machine interface using computers and improved information displays, improved surveillance and diagnostics regarding the condition of components such as computer-aided systems to provide early indication of sensor or component degradation, improved maintenance programmes and better operator qualification and simulator training. Other examples of advances applied at current plants to improve performance include high burnup fuel which supports longer cycle length, new materials for steam generators which have superior corrosion resistance and are now used for new and replacement PWR steam generators, and simpler systems for control of hydrogen (systems that require considerably less testing and maintenance and thereby reduce outage duration).

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, greater plant standardization and an overall goal of simplicity will contribute to high availability. Improved availability will also be gained by increased design margins that provide the capability to accommodate disturbances and transients without causing reactor trip, and provide additional assurance that plant lifetimes of 60 years can be achieved. Furthermore, the advances being applied at current plants that are mentioned above are also being incorporated into designs of future plants to contribute to high performance.

Technological advances are being incorporated to meet increasingly stringent safety objectives, by improving accident prevention as well as mitigation (see **Section 3.1**). Many of the advances have been tested to demonstrate technology readiness.

Economic competitiveness

Studies on projected costs of generating electricity provide results that depend strongly on the assumptions used. Due to the range of market conditions and generating costs in various countries, and the wide variety of assumptions used to forecast such costs, no single technology can be declared optimal in all countries. Importantly, in addition to economics, a country's national policy issues, such

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[7] 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 that would have been produced at maximum capacity under continuous operation during the reference period, and EL is the unavailable electrical energy that 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).

as diversity and security of its energy supply as well as environmental policies, may affect the decision on whether or not to construct nuclear power plants.

It is also important to note that the different generating options also have different cost sensitivities. Because of high capital costs and long construction periods, nuclear power generation costs, and, to a somewhat lesser extent, coal power generation costs, are highly sensitive to discount rates. Generating costs for coal-fired plants vary with coal prices and with the level of pollution abatement required. Generating costs for gas-fired power plants are highly sensitive to gas prices, which account for a large proportion of total costs[8].

In summary, capital cost reductions are needed to secure or enhance the competitiveness of future nuclear power plants. To meet this competitiveness challenge, construction delays must be avoided, regulatory procedures and requirements must be stable, plant design must be well in hand before the start of construction, and construction and operations management personnel need to have high levels of competence. It is important to fully implement proven means of cost reduction, and to examine, develop and implement new approaches to reduce costs. The proven means and new approaches for reducing costs are addressed in Ref. [7], and are summarized in **Section 2.2.**

Financing arrangements for new plants

Total investment costs and financing arrangements for implementing new nuclear power plant projects are affected by: direct costs, not including interest during construction; construction time; and the cost of capital/money.

Although it is expected that new reactor designs, evolutionary or innovative, will reduce capital costs and construction time of nuclear power plants, total investment costs of nuclear units are likely to remain high as compared to alternatives, in particular combined cycle gas turbines. Therefore, financing is a key issue to be addressed before the successful implementation of a nuclear power project.

It should be stressed, however, that the burden of finding financing arrangements is not unique to nuclear power plants as most projects in the energy sector are capital intensive. Traditionally, recognising the importance of energy infrastructures for industrial and social development, most governments have supported investments in the energy sector through policy measures or financial assistance.

Past experience has shown that financing of nuclear power projects is feasible in a large number of industrialised and developing countries. To date, more than 500 nuclear power reactors have been successfully financed and built. Financing nuclear power plants generally has been facilitated by need for base load electricity at stable projected production costs, competitiveness of the nuclear option, stable regulatory regime and indirect or direct government support.

The trend to market deregulation and to privatisation creates new challenges for financing equipment in the energy sector, in particular large power plants for base load generation. In the case of nuclear power, the financial risks for private investors in deregulated markets arise not only from political and regulatory uncertainties during construction and operation but also from uncertainties on the volume of demand and market prices.

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[8] In this context it is important to note that liberalized markets do not necessarily favour less capital intensive energy conversion systems and penalize capital intensive projects. Under conditions of low power prices and increasing prices for fossil fuel, the capital investment payback times for nuclear plants can be shorter than those for coal fired plants and CCGT plants [18].

Clearly, large generators or holdings are in a better position to face the financial risks of nuclear power projects. The 5th nuclear power plant project in Finland illustrates this point: the investor TVO is a large consortium of energy intensive industries which has the capabilities to face financial risks and also limited uncertainties on future demand since the shareholders of the company are also the main consumers.

An illustration of the role of governments is provided by the considerations in the U.S. regarding government incentives such as, for example, production tax credits, accelerated depreciation for tax purposes or loan guarantees for construction of some new nuclear power plants in the near term.

The approach of some designers to reduce financial risks is to downsize reactors. Although going to smaller size reduces the benefit of economy of scale, it allows to take advantage of design simplicity, modularity and factory-based manufacturing, limits financial risks and facilitates adaptation to uncertainties in future demand.

Generally, nuclear power plants are financed by the conventional approach that consists of multi-source financing, where a complete financing package is put together, covering the entire cost of the project. The first source is the investor/owner/operator. Its own resources constitute the basis of the financing package. In addition, bond issues, domestic bank credits and, in case of state-owned or controlled enterprises, funding from the governmental budget, should complete the financing needed.

In case of importing a nuclear plant from abroad, the conventional approach to financing the imported portion of the project is to invite financed bids. Export credits typically form the basis of the foreign financing package, because these generally have the most favourable terms and conditions. Suppliers from several countries may join in a consortium subdividing the scope of supply and involving several export credit agencies. For imported projects, the portion of work performed by domestic companies and labour force should be financed locally, or incorporated into the financing package.

Nuclear power and sustainable development

Currently countries disagree about the role of nuclear energy in sustainable development. Those that consider nuclear energy incompatible with sustainable development argue that it is too expensive and the associated risks are too high. Those that consider nuclear energy an important contributor to sustainable development emphasize its potential for significantly expanding electricity supplies while producing virtually no air pollution or greenhouse gases. Advanced designs could exploit fuel resources lasting centuries, and could produce products beyond electricity such as potable water from seawater and hydrogen for fuel-cell vehicles and other applications. The IAEA assesses various issues that are at the center of this debate [19]. Globally, nuclear power currently avoids approximately 600 million tonnes of carbon emissions annually, amounting to about 8% of global greenhouse gas (GHG) emissions. Fair markets (without prejudice for or against any energy generating options), which internalize all environmental and life cycle costs associated with energy production and use, are considered to be the best way to assure cost effective GHG reductions. Nuclear power today has already internalized costs to a greater extent than alternative technologies. Given the advantages that many countries see in nuclear power as a contributor to their sustainable development goals, it should be an important part of the future energy mix in many countries.

The implementation of the Kyoto Protocol and the application of its "flexible mechanisms" are at the forefront of energy policy debates in several countries. Several issues are raised within the debate on the role of nuclear energy in sustainable development. According to their interests and priorities, stakeholders in governments and civil society in different countries can view the potential role of nuclear energy quite differently. Ref. [20] provides key facts concerning nuclear energy and the Kyoto Protocol, highlights the challenges and opportunities for the future development of nuclear energy in the context of implementing the Kyoto protocol, and more broadly in alleviating the risks of global climate change.

Public acceptance

Public opinion can have a strong influence on decisions to build new plants. Public opinion is never uniform, is country-dependent and can be influenced by national priorities. The news media also have a strong influence on public opinion. A strong focus is needed to take care to present balanced and objective information on nuclear power issues. Broadening the dialogue with all interested groups to disseminate information outside of the nuclear community is necessary to give nuclear power a fair hearing.

Discussions of potential new nuclear energy projects as well as continued operation of existing plants often raises social concerns about risks of potential release of radioactivity, radioactive waste disposal and other issues associated with nuclear power. Establishment of the best energy policies from the viewpoint of society requires understanding and agreement between civil society and decision makers on issues related to nuclear energy. The relevance of nuclear energy in protecting the environment and contributing to sustainable energy strategies implies the importance of addressing issues that challenge the future development of nuclear energy including social perception of the risks associated with nuclear energy. It is highly important to understand the views of civil society on nuclear technologies, how the risks are perceived, and how to establish effective communication between all stakeholders aiming at enhancing consensus building prior to decision making. In that context, a recent study by the OECD-NEA [21] investigated various issues associated with nuclear energy and highlights the importance of risk perception and communication.

To help regain and maintain public acceptance wherever it has been lost, the nuclear community needs to focus on (a) maintaining a high level of safety at operating plants; (b) further development of technologies for future plants for assuring a high degree of safety, including severe accident prevention and mitigation measures; (c) demonstrating and implementing high level waste disposal; and (d) describing nuclear safety to the public in clear terms which express the benefits of concentrating and disposing of nuclear waste in contrast with the approach of dilution and dispersion of waste used by technologies which burn fossil fuels.

1.4. Non-electric applications of nuclear energy

As has been discussed in the preceding sections, nuclear energy is playing an important role in electricity generation, producing about 16% of the world's electricity. However, only about one-fifth of the world's energy consumption is used for electricity generation [22]. Most of the world's energy consumption is for heat and transportation. There is currently some use of nuclear energy for providing heat, and interest in the future use of nuclear energy in the heat market is growing. Nuclear energy has considerable potential to penetrate into the energy sectors now served by fossil fuels.

For heat applications of nuclear energy, the temperature requirements vary greatly. As shown in Figure 5 from [23], for heat applications the temperatures range from around 100°C for hot water and steam for district heating and seawater desalination, to up to 1000°C for process heat for the production of hydrogen by high temperature thermo-chemical processes. Although various forms of nuclear heat application are technically feasible and pursued between these temperature ranges, the major applications are directed to the lower end using water cooled reactors [24] and to the higher end using high temperature gas cooled reactors [23], [25], [26]. Figure 5 also shows the temperatures produced by the various reactor types[9] and the temperatures necessary for different non-electric applications.

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[9] It should be noted that reactor and fuel technology development could increase the achievable temperatures. For example, super-critical water cooled reactors would provide temperatures up to about 500°C; a lead or lead-bismuth cooled fast reactor system possibly may achieve core outlet temperatures ranging up to 800°C with advanced materials; and a graphite moderated, helium cooled, very high temperature reactor system would supply heat with core outlet temperature above 1000°C [27].

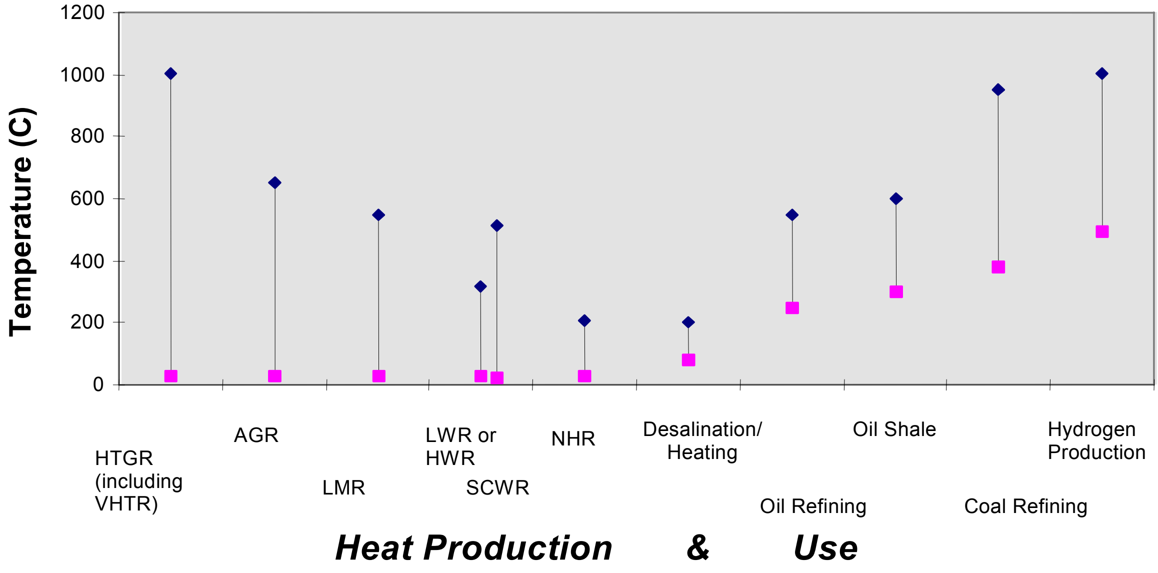

FIG. 5. Temperatures of heat produced by different reactor types and temperatures of heat used for different non-electric applications of nuclear energy. See text for an explanation of terms.

Low temperature heat applications include district heating, seawater desalination and a large variety of agricultural and industrial processes. Seawater desalination requires temperatures up to about 130 °C, district heating up to about 170 °C and low temperature industrial processes up to about 250 °C. Applications involving use of high temperature nuclear heat are not well proven and remain in the laboratory or in small scale demonstration phase. For large scale deployment significant research and development is still required.

Nuclear energy for hydrogen production

Hydrogen as an energy carrier is receiving increasing attention in OECD countries, notably in the U.S. and the European Union. Ref. [28] examines the wide range of activities required to realize hydrogen's potential in solving U.S. energy security, diversity, and environmental needs. Ref. [29] provides a vision outlining the research, deployment and non-technical actions that would be necessary to move from today's fossil-based energy economy to a future sustainable hydrogen-oriented economy with fuel cell energy converters.

Nuclear energy can be used for hydrogen production by using nuclear produced electricity for water electrolysis or by using nuclear heat from high or very high temperature reactors for indirect thermo-chemical water–splitting cycles. Production of hydrogen by nuclear electricity and / or high temperature nuclear heat would open the application of nuclear energy for the transportation sector and reduce the reliance of the transportation sector on fossil fuel with the associated price volatility, finite supply and greenhouse gas emissions. Using electricity in electrolyzers to produce hydrogen would allow a near-term option for distributed hydrogen generation at the point of delivery to the customer, such as at a fuelling station. Although the efficiency of hydrogen production by electrolyzers is lower than with high temperature thermo-chemical processes, such distributed production could play an initiating role, because of the lower capital investment and especially until large networks for hydrogen distribution become common. In the longer term, production of hydrogen at central nuclear stations with high or very high temperature reactors connected to extensive distribution networks may become cost efficient, with distributed production continuing to meet some needs.

Some experience for high temperature applications of nuclear energy is available on the laboratory scale and from component tests for earlier development programmes for HTGR applications. Significant research and development is still required before large scale deployment such as steam reforming of methane and thermo-chemical cycles for production of hydrogen.

Programmes are on-going in Japan and China with the goal of demonstrating the use of heat from HTGRs for high temperature applications [30] and [31]. In the USA, construction of an advanced reactor for hydrogen production is under consideration.

District heating

District heating networks generally have installed capacities in the range of 600 to 1200 MW(th) in large cities, decreasing to approximately 10 to 50 MW(th) in towns and small communities. For heat applications with nuclear plants, there are basically two options: Co-generation of electricity and heat, and dedicated nuclear heating reactors. Co-generation has been widely applied and experienced. In the co-generation mode, electricity will usually constitute the main product. Large size reactors, therefore, have to be integrated into the electrical grid system and optimized for base load electricity production. For reactors in the small to medium-size range, and in particular for small and very small reactors, the share of process heat generation could be larger, and heat could even be the predominant product.

Experience with nuclear district heating has been gained in Bulgaria, China, Germany, Hungary, Russia, Slovakia, Sweden, Switzerland and the Ukraine. A listing of operating nuclear heating plants is provided in [23]. Obviously, a potential market for the application of nuclear energy for district heating appears mainly in climatic zones with relatively long and cold winters. In Western Europe, for example Finland, Sweden, and Denmark are countries where district heating is widely used.

In the district heating field, the Russian Federation is reflecting its accumulated extensive experience in the improved design concept of a local district heating source and heat supply system. Restarting of the construction work of the site of Voronez and Tomsk is expected, both using AST-500 reactors. Some other cogeneration plants for district heating are also foreseen for replacing existing plants that are approaching the end of their design lifetime.

Seawater desalination

Application of nuclear heat for seawater desalination is another field with some operational experience and good prospects. Freshwater is essential in civilization and development. Its demand is rapidly growing throughout the world and some regions are already being jeopardized with the shortage of fresh water. Seawater desalination is a process of separating dissolved saline components from seawater to obtain fresh water with low salinity, adequate for irrigation, drinking and industrial use. Seawater desalination technologies[10] have been well established in the middle of the 20th century, with

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[10] These technologies can be classified as:

Multi-Stage Flash (MSF): MSF is a distillation process by which feed saline water (usually seawater) is allowed to flash along the lower sections of flash chambers (or stages), after the feed water has been heated in a primary vessel called the brine heater to temperatures in the range of 90–110^{o}C. Water vapor produced in the consecutive flashing stages is condensed in the upper sections on condensing tubes and collected on collection trays of the different stages as product distilled water. The concentrate brine reject is typically discharged to the sea.

Multi-Effect Distillation (MED): MED, similarly to MSF, takes place in a series of vessels (effects), where principles of condensation/evaporation and gradually reduced ambient pressure in down stream effects permit the seawater feed to undergo boiling without the need to supply additional heat after the first effect.

Reverse Osmosis (RO): RO is a membrane process in which the water from pressurized saline water feed is separated from the solutes (the salts) while passing through a semi-permeable membrane. No heating or phase change is necessary in this process since most of the energy required is for pressurizing the feed saline water.

still further improvement potential. The contracted capacity of desalination plants for desalinated water now exceeds 32 million m^3/d worldwide [32].

Nuclear desalination is the production of potable water from seawater in an integrated facility in which a nuclear reactor is used as the source of energy (electrical and/or thermal) for the desalination process on the same site. The facility may be dedicated solely to the production of potable water, or may be used for the generation of electricity and the production of potable water, in which case only a portion of the total energy output of the reactor is used for water production.

The experience and future opportunities for nuclear desalination were reviewed at a Symposium on Nuclear Desalination of Seawater, [33] convened by the IAEA in May 1997. Reference [26] summarizes global experience in nuclear seawater desalination and provides a list of operating nuclear desalination plants as of mid-2000.

The technical feasibility of integrated nuclear desalination has been firmly established by successful operation at several plants. This successful operation has proved the compliance with safety requirements and the reliability of co-generation nuclear reactors. Operating experience exceeds 150 reactor-years (statistics updated [23]).

Many IAEA Member States are moving forward in preparing nuclear desalination projects [34] and [35]. Activities are currently ongoing in Argentina, Canada, China, Egypt, France, India, Indonesia, Pakistan, the Republic of Korea, Morocco, the Russian Federation, and Tunisia.

In 1996 the IAEA, in its Options Identification Programme (OIP) identified practical options [36] of technical configuration of nuclear and desalination coupling to build near term technical and economic confidence under specific conditions. They were: (1) desalination in combination with a nuclear power reactor being constructed or in an advanced design stage with construction expected in the near term; (2) desalination, as above, in combination with a currently operating reactor with some minor design modifications as required to the periphery of the existing nuclear system; and (3) desalination in combination with a small (heating) reactor.

In 1997, the IAEA established the International Nuclear Desalination Advisory Group that provides a comprehensive and regular forum for the exchange of information on nuclear desalination technologies and programs. It also provides technical guidance for facilitating development of viable coupling configurations of nuclear and desalination systems.

In the Republic of Korea, the design of a nuclear desalination plant with the SMART reactor is developed to supply 40,000 m^3/day of fresh water and 90 MW of electricity to an area with an approximate population of 100,000 or an industrialized complex. A detailed design and construction project of a one-fifth scale SMART Pilot plant for demonstration of the relevant technologies, is currently underway and will be completed by 2008. This is an example of option 1 identified in the OIP. Also in Russia, efforts continue on a floating power unit based on a KLT-40 reactor for multipurpose use including desalination. A nuclear desalination project is foreseen in the Russian Arctic Sea coast area (Severodvinsk or Pevec) using an RO and/or MED process.

For option 2, three examples can be mentioned. A small RO facility set up at the KANUPP HWR unit in Pakistan has been in service since early 2000 producing 450 m^3/day of fresh water, and work is progressing on a Desalination Demonstration Plant, to be commissioned in 2005 at KANUPP. A 6300 m^3/day Multi-Stage Flash-Reverse Osmosis hybrid desalination plant is in commissioning in India at the Kalpakkam nuclear power plant. The product water is both for process water for the nuclear power plant and for drinking water in the neighbouring community. The Reverse Osmosis plant segment at Kalpakkam has been operating since 2002.

In Europe, the European Commission's project EURODESAL, coordinated by the French CEA, with partners in Europe and Canada, has conducted feasibility studies for both option (1) and option (2) above.

In addition to these activities, preheat Reverse Osmosis desalination experimental facilities are being set up in Egypt and Canada. Other countries are assessing a possibility of nuclear desalination plant under different time frame. For example,

- Egypt is continuing its feasibility study for an electricity and desalination plant at El-Dabaa.
- Tunisia is about to collaborate with France on a feasibility study of nuclear desalination for a site (la Skhira) in the southern part of the country along the Mediterranean coast.
- Indonesia is starting a joint feasibility study with Republic of Korea of a nuclear desalination plant in its Madura Island.
- In China, a nuclear desalination plant, based on the 200 MW(th) nuclear heating reactor with a capacity of 150,000 m^3/day is being studied for YanTai in Shandong Province.

As any nuclear reactor can provide energy (low-grade heat and/or electricity), as required by desalination processes, in principle, a broad option of coupling configurations can be feasible for future deployment of nuclear desalination.

Heat for other industrial processes

Within the industrial sector, at temperatures higher than those needed for district heating and seawater desalination, process heat is used for a variety of applications. Heat applications at temperatures up to about 200 to 300°C include the pulp and paper industry and the textile industry. Chemical industries, oil refining, oil shale and oil-sand processing and coal gasification are examples of industries with temperature requirements up to the 500–600°C level. Refinement of coal and lignite, and hydrogen production by water splitting, are among applications that require temperatures of 600–1000°C and above. Unlike district heating, the load factors of industrial users of heat do not depend on climatic conditions. The demands of large industrial users usually have base load characteristics.

Experience with provision of process steam for industrial purposes with nuclear energy has been gained in Canada, Germany, Norway and Switzerland. In Canada, steam from the Bruce Nuclear Power Development (BNPD) was supplied until the mid-to-late 1990s to heavy water production plants and to an adjacent industrial park at the Bruce Energy Center.

In Germany, since December 1983, the Stade PWR, has supplied steam for a salt refinery that is located at a distance of 1.5 km. In Norway, the Halden Reactor has supplied steam to a nearby factory for many years. In Switzerland, since 1979, the Gösgen PWR has provided process steam for a nearby cardboard factory.

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CHAPTER 2. TRENDS IN ADVANCED LWR DESIGN AND TECHNOLOGY

2.1. Development of advanced LWR designs

LWR plants offer a broadly developed and mature technology basis, but there is still a potential for further improvement; history shows that technology advances and entirely new knowledge, as well as operating experience, provide the basis for continual stepwise improvements of plant or equipment designs in all areas of industrial activity. Therefore, with time, nuclear power plant designs will inevitably change, with designers and utilities pursuing the best possible product through an emphasis on improved economy, reliability, availability, and safety, which represent common themes for new designs.

A list of the advanced light water reactor designs presented and discussed in this document including type, size, design organization and status is provided in Table I.

TABLE I. ADVANCED LIGHT WATER REACTOR DESIGNS

A) Large size advanced LWR designs (700 MW(e) or larger)

Name	Type	MW(e) Gross	MW(e) Net	Design Organizations	Status
ABWR	BWR	1385	1300	General Electric, USA; Hitachi Ltd. and Toshiba Corp., Japan	Operating in Japan Under construction in Japan and Taiwan, China Design certified by the U.S.NRC in USA
ABWR-II	BWR	1717	1638	Japanese utilities, General Electric, Hitachi Ltd. and Toshiba Corp.	In design phase – commercial introduction foreseen in latter half of 2010s
APWR	PWR	1538	-----	Mitsubishi, Japan/Westinghouse, USA	First units planned at the Japan Atomic Power Company's Tsuruga-3 and 4.
APWR+	PWR	1750	-----	Mitsubishi, Japan	In design phase – target for starting construction of a first unit is the end of the 2010s.
BWR 90+	BWR	----	1575	Westinghouse Atom, Sweden	Plant design is essentially complete
EPR	PWR	1650	~1550	Framatome ANP France/Germany	Detailed design completed
ESBWR	BWR	1390	1333	General Electric, USA	The Design Certification Pre-application review by the U.S.NRC was initiated in 2002
KSNP+	PWR	1050	1000	Korea Hydro and Nuclear Power Company, Republic of Korea	First units planned at KHNP's Shin-Kori-1 and 2
APR-1400	PWR	1450	1400	Korea Hydro and Nuclear Power Company etc., Republic of Korea	First units planned at KHNP's Shin-Kori-3 and 4
AP-1000	PWR	1200	1117	Westinghouse, USA	Under review by the U.S.NRC for Design Certification

Name	Type	MW(e) <u>Gross</u>	MW(e) <u>Net</u>	Design Organizations	Status
EP-1000	PWR	(see values for AP-1000)		Westinghouse, USA/Genesi, Italy	Programme now merged with AP-1000 programme. Design and analyses are being conducted to document compliance with European Utility Requirements
SWR 1000	BWR	1290	1250	Framatome ANP, Germany	In the U.S., the Design Certification Pre-application review by the U.S.NRC was initiated in 2002
WWER-1000 (V-392)	PWR	1068	1000	Atomenergoproject/Gidropress, Russia	Design is licensed for Novovoronezh Phase 2 (units 5 & 6) in Russia. The main design features were used for the two WWER units under construction at Kudankulam in India
WWER-1000 (V-466)	PWR	----[a)]	-----	Gidropress, Russia	Reactor plant design is developed for WWER-91/99, NPP92 and Balakovo-5 NPPs
WWER-1500 (V-448)	PWR	----[b)]	-----	Gidropress, Russia	Detailed design of reactor plant is under development
CNP1000	PWR	1000	-----	China National Nuclear Corporation, China	Engineering design
SCPR	SCWR	950	----	Toshiba, et al., Japan	Representative of Super-Critical Water cooled Reactor system selected by the Generation-IV International Forum

a) Thermal power is 3000 MW

b) Thermal power is 4250 MW

RMWR[c)]	BWR	1356	1300	JAERI, Japan	Design studies and experiments being performed. Small scale prototype possible by early 2010s; commercialization by 2020
RBWR	BWR	----	1356	Hitachi, Japan	design studies

c) A small scale (300 MW(e) class) RMWR with passive safety features is also being developed by JAERI, JAPC, Hitachi and Tokyo Institute of Technology under the innovative and viable nuclear energy technology development program (IVNET) sponsored by the Ministry of Economy, Trade and Industries (METI) of Japan since FY2000

TABLE I. ADVANCED LIGHT WATER REACTOR DESIGNS

B) Medium size advanced LWR designs (300-700 MW(e))

Name	Type	MW(e) Gross	MW(e) Net	Design Organizations	Status
AC-600	PWR	600	----	China National Nuclear Corporation, China	R&D results will be applied to development of large advanced PWR
AP-600	PWR	619	600	Westinghouse, USA	Design has been certified by the U.S. Nuclear Regulatory Commission
HSBWR	BWR	600	----	Hitachi, Japan	Conceptual design
HABWR	BWR	650	----	Hitachi, Japan	Conceptual design
WWER-640 (V-407)	PWR	640	----	Atomenergoproject, St. Petersburg / Gidropress, Russian Federation	Construction of pilot plant at Sosnovy Bor site is under consideration. This would be followed by units at the Kola nuclear power station and other sites.
VK-300	BWR	---	2x250[d)]	RDIPE, Russian Federation	Design. Testing of key systems and components underway

d) A twin unit VK-300 electrical plant would produce 2 x 250 MW(e). The VK-300 may be used for co-generation of district heat and electricity (at a reduced electrical capacity rating).

IRIS	PWR	----	335	Westinghouse, USA	In Pre-application Review for Design Certification by the U.S.NRC. Westinghouse expects that IRIS will be submitted to the U.S. NRC for Design Certification in 2004/5, with Design Certification following in 2008/2009.
QS-600e/w Co-generation plant	PWR	644	610[e)]	CNNC, China	Conceptual design
PAES-600 with twin VBER-300 units	PWR	---	2x295	OKBM, Russian Federation	Conceptual design
IMR	PWR	330	----	Mitsubishi, Japan	Conceptual design
NP-300	PWR	334	314	Technicatome, France	Basic design

e) This is the net electric rating for a plant that produces only electricity with no heat for desalination. A co-generation plant used for sea-water desalination and electric power production would have a lower net electric power capacity.

TABLE I. ADVANCED LIGHT WATER REACTOR DESIGNS

C) Small size advanced LWR designs (below 300 MW(e))

Name	**Type**	**MW(e) Gross**	**MW(e) Net**	**Design Organizations**	**Status**
LSBWR	BWR	306	----	Toshiba, Japan	Conceptual design
CAREM	PWR	----[f)]	27[g)]	CNEA/INVAP, Argentina	Conceptual engineering for 27 MW(e) prototype, which is under consideration, has been completed
SMART	PWR	90	----[h)]	KAERI, Republic of Korea	Design and construction project for a 1/5th scale pilot plant is under way with completion planned by 2008
SSBWR	BWR	150	-----	Hitachi, Japan	Conceptual design
KLT-40	PWR	----	up to 70[i)]	OKBM, Russian Federation	A first unit, an adaptation of a nuclear propulsion unit used for the ice-breaker fleet in Russia, is planned at Severodvinsk of the Arkhangelsk region in the Russian Federation.
PSRD-100	PWR	----	31[j)]	JAERI, Japan	Conceptual design

f) CAREM concepts are in the small size range, utilizing natural circulation for plants below 150 MW(e), or forced flow for plants with larger ratings.

g) Rating of prototype.

h) The thermal power of the full sized unit is 330 MW, to be used in the co-generation mode for 90 MW(e) (gross) of electric power and for sea-water desalination to produce 40,000 m^3 of fresh water per day.

i) Depending on amount of heat used in co-generating mode.

j) The concept reported in this TECDOC is rated 100 MWt. A 300 MWt concept is also being developed.

As discussed in the Introduction, the development activities for advanced plants can be divided into two categories, based on the degree of deviation from existing designs, or rather the need for confirmation and validation before commercial deployment. Evolutionary designs represent descendants from existing plant designs featuring improvements and modifications based on feedback of operating experience and/or aimed at adopting new technological developments. The other category covers design concepts that will incorporate more significant departures from existing nuclear power plant designs, and will require much more development effort, possibly including construction and operation of a prototype and/or demonstration plant before their commercial deployment. Designs in this category that incorporate radical conceptual changes in design approaches or system configuration in comparison with established practice are often referred to as innovative designs.

Some of the evolutionary designs build closely on earlier designs and need only engineering efforts before a commercial deployment. Other evolutionary designs incorporate more new features and depart more from existing designs, e.g. by introduction of passive safety features to replace, or supplement, traditional safety features and systems to achieve plant simplification. For such designs, confirmatory testing of the new features and components, in addition to the engineering efforts, are conducted prior to the commercial deployment.

From the history of nuclear power plant development, it becomes evident that times have changed. In the past, designs, and basic design requirements, were often developed by the different vendor organizations themselves, with regulatory requirements serving as the basis; specific utility requirements would very much be focused on local needs and conditions. Today, the situation is different; there is a clear tendency among both vendors and utilities to think more broadly and require designs that would be suitable for deployment in many countries. The reasons are quite clearly economical concerns; a standardized design minimizes engineering needs, facilitates use of well-qualified equipment, simplifies spare parts schemes, and enables efficient co-operation on plant operation procedures and problem solving. Hence, quite a number of international co-operative efforts have been established, aiming at establishing “standardized solutions”, on the supplier side, on the utility side, and on the regulatory side.

With respect to the latter, the efforts by the utilities, and by the regulators, are very important, and in this context, the Utility Requirements Document (URD) that has been developed by the Electric Power Research Institute (EPRI) and the European Utility Requirements (EUR) document that has been developed by the major European utilities are addressed to design organizations to provide guidance for new designs. These and other utility requirements documents are discussed in Chapter 3.

Overview of evolutionary LWR development

In France and Germany, Framatome ANP has completed the basic design for the large size European Pressurized Water Reactor (EPR) in 1998, which meets European utility requirements. The EPR’s higher power level relative to the latest series of PWRs operating in France (the N4 series) and Germany (the Konvoi series) has been selected to capture economies of scale.

In Germany, Framatome ANP with international partners from Finland, the Netherlands, Switzerland and France is developing the basic design of the SWR-1000, an advanced BWR with passive safety features.

In Japan, benefits of standardization and construction in series are being realized with the ABWR units[11]. Expectations are that future ABWRs will achieve a significant reduction in generation cost relative to the first ABWRs. The means for achieving this cost reduction include standardization, design changes and improvement of project management, with all areas building on the experience of the ABWRs currently in operation. In addition, a development programme was started in 1991 for

[11] The first two ABWRs in Japan, the 1360 MW(e) Kashiwazaki-Kariwa 6 and 7 units, have been in commercial operation since 1996 and 1997 respectively. ABWR plants are under construction at Hamaoka Unit No. 5 and Shika Unit No. 2.

ABWR-II, aiming to further improve and evolve the ABWR, with the goal of significant reduction in power generation costs relative to a standardized ABWR. The power level of ABWR-II has been increased relative to the ABWR, and benefits of economies of scale are expected. Commissioning of the first ABWR-II is foreseen in the late 2010s. Also in Japan, the basic design of a large advanced PWR has been completed by Mitsubishi Heavy Industries and Westinghouse for the Japan Atomic Power Company's Tsuruga-3 and –4 units, and a larger version, the APWR$^+$ is in the design stage.

In the Republic of Korea, the benefits of standardization and construction in series are being realized with the Korean Standard Nuclear Plants (KSNPs)[12]. The accumulated experience is now being used by KEPCO to develop the improved KSNP$^+$. In addition, the development of the Korean Next Generation Reactor, now named the Advanced Power Reactor 1400 (APR-1400), was started in 1992, building on the experience of the KSNPs. Recent development of the APR-1400 focused on improving availability and reducing costs. The higher power level of the APR-1400 relative to the KSNP and the KSNP$^+$ has been selected to capture economies-of-scale. In March 2001, KEPCO started the Shin-kori 3,4 project for the APR1400.

In the USA, designs for a large sized advanced PWR (the Combustion Engineering System 80+) and a large sized BWR (General Electric's ABWR) were certified in May 1997. Westinghouse's mid-size AP-600 design with passive safety systems was certified in December 1999. Westinghouse is developing the AP-1000 applying the passive safety technology developed for the AP-600 with the goal of reducing the capital costs through economies-of-scale. The AP-1000 is under review by the U.S. NRC for Design Certification. An adaptation of the AP-1000, called the EP-1000, is being designed by Westinghouse and Genesi (Italy) applying the passive safety technology to meet European Utility Requirements and licensing requirements in Europe. A Westinghouse led international team is developing the modular, integral IRIS design in the small to medium-size range, with a core design capable of operating on a 4-year fuel cycle[13]. General Electric is designing a large ESBWR applying economies of scale together with modular passive safety systems. The design draws on technology features from General Electric's ABWR and from their earlier mid-size simplified BWR with passive systems.

In Sweden, Westinghouse Atom has developed the large BWR 90+, an advanced boiling water reactor with improved safety and operability.

In the Russian Federation, efforts continue on evolutionary versions of the currently operating WWER-1000 (V-320) plants. This includes the WWER-1000 (V-392) design, of which two units are planned at the Novovoronezh site, and WWER-1000 units under construction in China, India and the Islamic Republic of Iran. Development of a larger WWER-1500 design has been initiated.

In China, the China National Nuclear Corporation (CNNC) has developed the AC-600 design, and is currently developing the CNP-1000 for electricity production. CNNC is also developing the QS-600 e/w, which is based on the design of the Qinshan Phase II, for electricity production and sea water desalination. China is pursuing self-reliance both in designing the plant to meet Chinese safety requirements, and in fostering local equipment manufacture with the objective of reducing construction and operation costs. Experience gained and lessons learned from the design, construction and operation of the Qinshan and Daya Bay NPPs are being incorporated.

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[12] The first two KSNPs, Ulchin 3 and 4 began commercial operation in 1998 and 1999. Yonggwang 5 and 6 began commercial operation in 2002. Two more KSNPs are under construction at Ulchin 5 and 6. The first two units of KSNP$^+$ are planned at Shin Kori 1 and 2.

[13] IRIS is considered to be an evolutionary LWR in the context of Figure 1 in the Introduction to this TECDOC. IRIS has innovative features and the integral design represents a radical change in system configuration from existing loop reactors. However Westinghouse states that while it is innovative engineering, it relies on proven LWR technology and thus it only requires engineering and confirmatory testing. A prototype or demonstration plant is not required, but a first of a kind will be, since no other IRIS-type integral reactors have been built.

Overview of innovative LWR development

A trend in the design of small and medium sized light water reactors has been simplified designs with long core life and modular design for factory production of standardized components and systems. Several small to medium sized PWR designs are of the integral reactor type in which the steam generator is housed in the same vessel as the reactor core. This approach eliminates primary system piping. The Argentinian CAREM reactor (prototype design 27 MW(e)) is cooled by natural circulation, and has passive safety systems. The SMART design that has been developed in the Republic of Korea is an integral PWR and, like CAREM, uses no soluble boron. A decision has been made to build a 1/5th scale, 65 MW(th), SMART pilot plant. The Japan Atomic Energy Research Institute is developing the small passively safe integral PSRD-100 system for electricity and/or heat supply and seawater desalination, and Mitsubishi together with other organizations is developing the integral IMR design for electricity production.

In Russia, development is on-going at OKBM for both the VBER-300 integral design with the steam generator system inside the reactor pressure vessel and for the KLT-40, a floating small NPP design for electricity and heat; at RDIPE for the VK-300 BWR design for electricity and district heating; and at Atomenergoproject/Gidropress on a mid-size WWER-640 with passive safety systems.

In Japan, the Toshiba Corporation and the Tokyo Institute of Technology are developing a long operating cycle, natural circulation simplified LSBWR with passive safety systems. The LSBWR's power level is in the small size range with a target 15-year core life. Hitachi Ltd. is also developing the mid-size Hitachi Simplified BWR (HSBWR), the mid-size Advanced BWR (HABWR), and the small size SSBWR with passive safety systems and a 20-year core life.

Also in Japan, with the goals of ensuring sustainable energy supplies by achieving a high conversion ratio (equal to or beyond 1.0) and reducing spent fuel accumulation, Hitachi Ltd. is also developing the large size, reduced moderation RBWR and JAERI is developing the large size RMWR.

As is noted in the Introduction, supercritical water cooled reactor systems are one of the six systems that have been selected by the GIF. In a supercritical system the reactor operates above the critical point of water (22.4 MPa and 374 °C) resulting in higher thermal efficiency than current LWRs. Thermal efficiencies of 40–45% are projected with simplified plant designs. Core design options include both thermal neutron spectrum cores and fast neutron spectrum cores for high conversion. The large size SCPR concept of Toshiba and Japanese partners is presented in this document as an example supercritical water cooled reactor system. In Europe, the HP-LWR project has been funded by the European Commission to assess the merit and economic feasibility of an LWR operating thermodynamically in the supercritical regime. Activities on super-critical water cooled system concepts are also on-going at universities and research centers in the USA.

2.2. Summary of means for reducing costs and construction times

Proven means for reducing costs and construction times

There is a set of proven means for reducing costs during any construction project, including nuclear projects. Several studies [1], [2], [3] and [4] have addressed these means, which can be generally grouped and listed as follows:

1. Capturing economies-of-scale;
2. Streamlining construction methods;
3. Shortening construction period;
4. Standardization, and construction in series;
5. Multiple unit construction;
6. Simplifying plant design, improving plant arrangement, and use of modeling;
7. Efficient procurement and contracting;
8. Cost and quality control;

9. Efficient project management; and
10. Working closely and co-operating with relevant regulatory authorities.

This list has not changed much over time. The larger the construction project, and the greater the financing burden, as is the case for nuclear power plants, the more important these approaches become.

The best combination of approaches depends on market conditions. As discussed in **Section 2.1** in the sub-section on overview of evolutionary LWR development, in some countries **economies of scale** are being pursued for new, large evolutionary LWRs.

However, for some market conditions, increasing plant size to capture **economies of scale**, would result in plants too large for the grid or for incremental demand. Designs for small and medium size reactors (SMRs) offer the opposite choice. Economy is being pursued by design simplification, and the use of modular, factory fabricated systems to reduce the field construction time. SMRs have the potential to capture **economies of series production** instead of economies of scale, if several units are constructed.

Reducing the construction period is important because of the interest and financing charges that accrue during this period without countervailing revenue. However, the objective is to reduce overall cost, which means an optimization. It would not be meaningful to reduce the overall schedule period if that would increase overall spending or incur later costs in a way that negates the savings in interest during construction. One way to reduce the schedule is to reduce on-site and tailor-made construction and emphasize instead the manufacture of modular units or systems. Besides improvements in construction method, other measures that could reduce the construction schedule include advanced engineering methods, and up-front engineering and licensing.

Significant improvements can be made in plant **design** and layout, and use of computer technology and modelling. Several simplifications have been made in the last decade including computer control, process information display, and other areas. Careful planning can result in improvements in plant arrangement and system accessibility, and in design features to facilitate decommissioning.

Standardization and construction in series offer significant cost savings by spreading fixed costs over several units built, and from productivity gains in equipment manufacturing, field engineering, and building construction. First of a kind reactor designs or plant components require detailed safety cases and licensing procedures, resulting in major expenditures before any revenue is realized. Standardization of a series is therefore a vitally important component of capital cost reduction. Standardization and construction in series offer reduced average licensing times and costs over the series. A detailed account of the lessons from the standardized plant design and construction programme in France is provided in Ref. [5]. Experience is being established within Japan's ABWR activities and the Republic of Korea's KSNP and KSNP+ activities.

Closely related is the cost–saving practice of **multiple unit construction** at a single site. The average cost for identical units on the same site can be about 15% or more lower than the cost of a single unit, with savings coming mostly in siting and licensing costs, site labour and common facilities. A good example of multiple unit construction are the 58 PWRs that are operating in France, which have been built as multiple units at 19 sites.

Many of the benefits of technology advances would be lost without some accompanying **regulatory reform** to accommodate change. These include greater regulatory certainty, more prioritization of regulatory requirements, streamlining of regulation to match streamlined engineering and designs, and more flexibility to accommodate technological innovation.

In developing countries, **furthering self-reliance**, and **enhancing local participation** in major projects are goals pursued by governments for a variety of policy reasons. Cost savings in any of several areas — materials and construction costs, foreign exchange costs, labor costs — may result.

Reducing the costs of technology transfer and relevant training are areas of emphasis for developing countries. In China, it is considered very important that favourable conditions for technology transfer and personnel training are provided with the help of industrialized countries so that a considerable portion of the work in fabricating the plant equipment and in plant construction can be done by organizations in the developing country. Because of the low cost of manpower, some materials and products can be made cheaper, with due assurance of quality. Experience in China is that the construction cost of the Qinshan-II plant (2 x 600 MW(e) units, the first unit achieving commercial operation in March, 2002) indicates that the cost of this plant is less than that for imported large size plants because of localization of design and provision of a large amount of the equipment by domestic organizations.

Reference [1] provides further examples of recent and present activities to incorporate the proven means discussed above. These and other traditional proven approaches should help to achieve cost competitiveness for new nuclear power plants. However, the nuclear community must continue to move forward in identifying and implementing new approaches for further reducing the costs of new nuclear plants.

Reference [1] discusses new approaches to reduce capital cost that should be developed and implemented in order to gain the greatest possible cost reductions. In summary, these are:

- Modularization, factory fabrication, and series production;
- Development of highly reliable components and systems, including "*smart*" (instrumented and monitored) components and methods for detecting incipient failures — to improve system reliability so that dependence on costly redundancy and diversity practices could be reduced. Development is also required to correlate signals from the "smart" components with reliability, and criteria must be developed for when to do maintenance and replacement;
- Further development of passive safety systems where the safety function can be met more cheaply than with active systems. This would include development of reliability models for passive systems. A discussion of factors regarding the application of passive safety systems is provided in Appendix 2;
- Development of computer based advanced technologies for design, procurement, manufacture, construction and maintenance with a focus on coordination of activities to reduce costs and schedules;
- Further development of Probabilistic Safety Analysis (PSA) methods and data bases to support plant simplification and to support examination of potential risk-informed regulatory requirements for new plants leading to more economical designs with very high safety levels. PSA assessments must (a) be capable of assessing the total risk including full power, low power, shutdown, fires and external events; (b) be capable of accounting for safety culture and human factors; (c) accurately account for ageing effects; and (d) include capability to quantify uncertainties. The challenge will be to establish PSA methods, including understanding of uncertainties in predicted results, to demonstrate that sufficient defense-in-depth, and sufficient balance among the various levels of defense-in-depth, can be achieved through simpler and cheaper technical solutions;
- Improvement of the technology base for eliminating over-design (i.e. improved understanding of thermo-hydraulic phenomena, more accurate data bases of thermo-hydraulic relationships and thermo-physical properties, better neutronic and thermo-hydraulic codes, and further code validation). The focus could be on removing the need to incorporate excessively large margins into the design simply for the purpose of allowing for limitations of calculational methodology and uncertain data.
- Reduction of number of components and materials requiring nuclear grade standards;
- Design for higher temperature (higher thermal efficiency);

- Design for multiple applications (e.g. co-generation of electricity and heat; sea water desalination); and
- Achieving international consensus regarding commonly acceptable safety requirements that would facilitate development of standardized designs which can be built in many countries without requiring significant re-design efforts.

REFERENCES TO CHAPTER 2

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Improving Economics and Safety of Water Cooled Reactors: Proven Means and New Approaches, IAEA-TECDOC-1290, Vienna (2002).

[2] ELECTRIC POWER RESEARCH INSTITUTE, Advanced Light Water Reactor Utility Requirements Document, Volume I, Rev. 1, December 1995; Vol. II and III, Rev. 8, (March 1999).

[3] OECD/NEA, Reduction of Capital Costs of Nuclear Power Plants, OECD (2000).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Objectives for the Development of Advanced Nuclear Plants, IAEA-TECDOC-682, Vienna (1993).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Evolutionary Water Cooled Reactors: Strategic Issues, Technologies and Economic Viability, Proceedings of a symposium held in Seoul, 30 November – 4 December 1998, IAEA-TECDOC-1117, Vienna (1999).

CHAPTER 3. SAFETY OBJECTIVES AND UTILITY REQUIREMENTS FOR ADVANCED LIGHT WATER REACTORS

3.1. IAEA safety goals and requirements

In the course of nuclear power development in the latter part of the twentieth century, there have been significant developments in technology for reactor safety. These include:

- advances in the application of PSA;
- introduction of more rigorous quality assurance programmes for plant design, licensing, construction and operation;
- increased attention to the effect of internal and external hazards – in particular the seismic design and qualification of buildings;
- major advances in fracture mechanics and non-destructive testing and inspection;
- increased emphasis on the man-machine interface including improved control room design, and plant design for ease of maintenance;
- rapid progress in the field of control and instrumentation – in particular, the introduction of micro-processors into the reactor protection system; and
- increased emphasis on prevention and mitigation of severe accidents.

The IAEA is authorized by its Statute to "establish or adopt, in consultation and where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety for protection of health and minimization of danger to life and property". Publications of regulatory nature by means of which the Agency establishes safety standards are issued in the IAEA's **Safety Standards Series**, which consists of the following categories:

Safety Fundamentals present basic objectives, concepts and principles of safety and protection in the development and application of nuclear energy for peaceful purposes;

Safety Requirements establish the requirements that must be met to ensure safety. These requirements, which are expressed as 'shall' statements, are governed by the objectives and principles presented in the Safety Fundamentals; and

Safety Guides recommend actions, conditions or procedures for meeting safety requirements. Recommendations in Safety Guides are expressed as 'should' statements, with the implication that it is necessary to take the measures recommended or equivalent alternative measures to comply with the requirements.

The Agency's safety standards are not legally binding on Member States but may be adopted by them, at their own discretion, for use in national regulations in respect of their own activities[19]. The standards are binding on the Agency for application in relation to its own operations and to operations assisted by the Agency.

Advice provided by the IAEA's International Nuclear Safety Advisory Group (INSAG), which serves as a forum for exchange of information in nuclear safety issues of international significance, is taken into account, and recommendations made by a number of international bodies including the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), the International

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[19] The Agency's safety standards cover safety in five areas:

- safety of nuclear facilities;
- radiation protection and safety of radiation sources;
- safe management of radioactive waste;
- safe transport of radioactive material; and
- general safety (cross-cutting themes).

Commission on Radiological Protection (ICRP) and the International Commission on Radiation Units and Measurements (ICRU) are used as a basis in the preparation and review of the Agency's safety standards.

Safety fundamentals for nuclear plants are provided in the IAEA document "The Safety of Nuclear Installations" [1]. This document states the **General Nuclear Safety Objective** "to protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards". This objective is supported by two complementary safety objectives dealing with radiation protection and technical aspects:

Radiation Protection Objective: "to ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accident".

Technical Safety Objective: "to take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low."

Technical aspects of safety including principles are discussed in Reference [1] for siting, design and construction, commissioning, operation and maintenance, and radioactive waste management and decommissioning.

In 2000 the Agency published the document "Safety of Nuclear Power Plants: Design" [2] which establishes nuclear plant safety design requirements applicable to safety functions and associated structures, systems and components, as well as to procedures important to nuclear plant safety. It recognizes that technology and scientific knowledge will continue to develop, and that nuclear safety is not a static entity; however, these requirements reflect the current consensus. They are expressed as 'shall' statements, and are governed by the objectives and principles in the Safety Fundamentals document. The Design Requirements document avoids statements regarding the measures that 'should' be taken to comply with the requirements. Rather, Safety Guides are published from time to time by the Agency to recommend measures for meeting the requirements, with the implication that either these measures, or equivalent alternative measures, 'should' be taken to comply with the requirements.

Discussions of the safety of future plants often involve different types of probabilistic safety criteria (PSC). PSC can be defined as *limits*, not to be exceeded, or as *targets*, *goals* or *objectives* (to strive for, but without the implication of unacceptability if the criteria are not met). PSC can be related to the core damage frequency (CDF), which is predicted by performing a level 1 PSA. Another type of PSC can be related to the large early release frequency (LERF) that would follow from severe core damage together with a major early failure of the containment. Use of LERF in PSC carries the implication that a late failure of the containment may be averted by accident management procedures, or mitigated by emergency response (e.g. evacuation of the public in the vicinity of the plant).

Discussions of PSC *targets* for CDF and large off-site-release have been provided for more than a decade in INSAG documents [3], [4], [5], [6]. In 1988, INSAG-3 stated "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 in 1992 with the statement 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".

In 1996 INSAG-10 noted 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. INSAG-10 noted that values that are much smaller than this would, it is generally assumed, be difficult to validate by methods and with operating experience currently available. INSAG-10, therefore, considers improved mitigation to be an essential complementary means to ensure public safety. INSAG-10 also stated the need to demonstrate that for accidents without core melt there will be no necessity for protective measures (evacuation or sheltering) for people living in the vicinity of the plant, and for severe accidents that are considered in the design, that only protective measures that are very limited in area and time would be needed (including restrictions in food consumption). In 1999, INSAG-12 (Revision 1 of INSAG-3), confirmed that the target frequency for CDF for existing nuclear power plants is below about 10^{-4} with severe accident management and mitigation measures reducing by a factor of at least 10 the probability of large off-site releases requiring short term off-site response. INSAG-12 continued by noting that for future plants, improved accident prevention (e.g. reduced common mode failures, reduced complexity, increased inspectability and maintainability, extended use of passive features, optimized human-machine interface, extended use of information technology) could lead to achievement of an improved CDF goal of not more than 10^{-5} per reactor-year. With regard to off-site release for future plants, INSAG-12 stated that an objective for future plants is "the practical elimination of accident sequences that could lead to large early radioactive releases, whereas severe accidents that could imply a late containment failure would be considered in the design process with realistic assumptions and best estimate analyses so that their consequences would necessitate only protective measures limited in area and in time".

From the Safety Standards Series and INSAG documents, a number of safety goals for future nuclear plants can be identified:

- a reduction in core damage frequency (CDF) relative to current plants;
- consideration of selected severe accidents in the design of the plants;
- ensuring that releases to the environment in the event of a severe accident are kept as low as practicable with the aim of providing a technical basis for simplification of emergency planning;
- reduction of the operator burden during an accident by an improved man-machine interface;
- the adoption of digital instrumentation and control; and
- the introduction of passive components and systems.

Technological advances are being incorporated into advanced designs to meet the stringent safety goals and objectives. Design features both to improve prevention of severe accidents involving core damage, as well as for mitigating their consequences are being incorporated. Considerable development has been carried out worldwide on new systems for heat removal during accidents. Progress has been made in containment design and in instrumentation and control systems.

To further reduce the probability of accidents and to mitigate their consequences, designers are adopting various technical measures. Examples are:

- larger water inventories (large pressurizers, large steam generators), lower power densities, negative reactivity coefficients to increase margins and grace periods thereby reducing system challenges;
- redundant and diverse safety systems with proven high reliability with improved physical separation between systems;
- passive cooling and condensing systems; and
- stronger containments large enough to withstand the pressure and temperatures from design basis accidents without fast acting pressure reduction systems, and with support systems to assure their integrity during severe accidents (for example, to control hydrogen concentrations). In some designs there is an outer second containment that provides protection against external events, and allows for detection and filtration of activity which potentially would leak from the inner containment.

A recent review of trends in the development of water cooled reactors [7] presented a set of severe accident challenges that are being 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 these phenomena are quite complex, there is in general sufficient understanding to allow designers to make choices for features to cope with them.

The advances in understanding of the phenomena are, in many cases, leading to common approaches being employed by designers to deal with the challenges. Early containment failure from 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 to prevent hydrogen combustion in PWRs involve large volume containments and installation of igniters and/or autocatalytic re-combiners to further reduce likelihood of hydrogen combustion. An alternate approach, which is incorporated into BWR designs, is to inert the containment atmosphere by means of nitrogen. This eliminates the potential for destructive fires and prevents hydrogen combustion.

In other areas, such as steam explosions and debris coolability, research has improved the understanding of the phenomena but designers are adopting different strategies to prevent or mitigate the challenge to the containment. In the area of 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 also different 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 from below by water flowing in tubes imbedded in the concrete.

Some new plant designs rely on redundant and diverse active safety systems to transfer decay heat from the primary system and finally from the containment building during accidents. A high degree of reliability and safety with active systems can be achieved through redundancy, separation, and diversity, and by assuring with high confidence the supply of electric power for their operation. Other new designs incorporate safety systems that rely on passive means such as gravity, natural circulation, and compressed gas as driving forces, to transfer heat to either evaporating water pools or to structures cooled by air convection. Passive systems are considered in the design process as a means of simplifying safety systems and thereby reducing cost, improving reliability, mitigating the effect of human errors and equipment failures, increasing the time operators have available to cope with accident conditions, and reducing reliance on off-site and on-site power supplies. However, passive systems have lower driving forces and less operational flexibility.

In some designs a coupling of active safety systems and passive safety systems is adopted. The balance between active and passive systems is plant-specific and must take into account validation for plant conditions, integration into the overall plant safety systems, in-service inspection requirements, maintenance, reliability and the impact on costs. The main requirement is that the proposed system fulfils the necessary function with appropriate reliability. Passive systems have an advantage in areas that can be contaminated in an accident, since such areas may be inaccessible for repair. Adequate testing of passive systems is important to determine conditions that affect 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 the last barrier to prevent large releases of radioactive material in the event of an accident. The types of containment designs are in the categories of pre-stressed or reinforced single concrete containments with a steel liner; cylindrical and spherical steel containments; and pre-stressed double containment with and without a steel liner. The early smaller designs used steel containments, but for larger designs, the requirements on steel containments have become difficult to satisfy and concrete containments are more common. The containment design enhancements have basically been mentioned above as measures to mitigate severe accidents, and include higher design pressure, low leakage factors, as well as measures to protect the containment including reactor cavity flooding system, hydrogen control systems, means for spreading and cooling a molten core, and sometimes a double containment as discussed earlier.

The development of electronics, computers and software, and instrumentation and control technology is progressing rapidly, offering opportunities to enhance the safety of nuclear plants. Because there have been few orders for new reactors in the past decade, the nuclear industry has been slow to take advantage of these developments. However, the situation is now changing as equipment in current plants is becoming obsolete and is being replaced, and experience with new I&C systems is now being obtained through implementation of advanced systems in modernization projects for current plants. For the evolutionary water cooled reactor designs, modern instrumentation and control equipment is a fully integrated feature.

Importantly, design measures both for increased prevention as well as for accident mitigation tend to increase capital cost, although preventive measures (such as increased margins) 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. Therefore there is a strong connection between the continued efforts focusing on achieving economical nuclear power in an increasingly competitive environment, and approaches to achieve more stringent safety targets. Clearly, it is important to pursue technological advances both for improving economics and for improving safety.

3.2. Utility Requirements for Advanced LWRs

3.2.1. Overview of utility requirements documents

New nuclear plant designs must be profitable for the electricity producer, acceptable by the regulatory bodies and attractive to the public.

During the first decades of introduction of nuclear power into several countries, the buyors/operators of nuclear power plants, basically utilities or groups of utilities, relied on the advice and experience of the nuclear plant suppliers for deliveries of “turnkey” nuclear power plants or nuclear steam supply systems (NSSS) for such plants. The plant designs were established by the suppliers and architect engineers based on their experience, and the utilities mainly limited their involvement in the design and construction to certain review work and to training on the job. The role of the operating utility included the responsibility for plant safety vis-a-vis the regulatory authorities. The licensing of nuclear power plants normally followed internationally established rules taking into account also national codes and standards of the country intending to build the plant. When specific national codes did not exist, however, the codes and standards of the country of origin were frequently taken as the basis.

By the mid 1970s, several operators had accumulated a lot of experience from their operating nuclear power plants, some good and others less good. New or modified requirements that the plant had to be adapted to were often imposed during the lifetime of the plant; and the adaptations sometimes required certain re-engineering as well as new design approaches. The utilities started to specify their special desires for new nuclear plants, reflecting their specific experience, in bid invitation specifications. These can be seen as the first steps to the development of utility requirements for future advanced reactor concepts. The utilities realized that it would be beneficial to bring together all the experience gained, and established so called owners’ groups (e.g., BWR and PWR owners’ groups) in order to exchange operating experience with a specific type of reactor.

Based on the large accumulated experience, utilities began to indicate desires for a new generation of standardized plant designs that should be more economically competitive, safer, simpler and have greater design margins. This trend soon evolved into extensive activities to develop requirements for standardized plant designs based on the operating experience and accounting for advances in research and development. For example, in the mid 1970s, the ministry of international trade and industry (MITI) in Japan initiated a long term “LWR Standardization Program for Improvements”, and in the

United States, a major utility project was launched in the mid 1980s to establish a complete set of utility requirements for a next generation of standardized nuclear power plants.

3.2.1.1. EPRI Utility Requirements Document (URD)

Beginning in 1985, U.S. utilities led an industry-wide effort to utilize the accumulated knowledge and experience for the development of technical bases for the design of advanced light water reactors (ALWRs). The effort was managed for the U.S. electric industry by the Electric Power Research Institute (EPRI), in close co-operation with the U.S. Department of Energy (DOE), and it included participation and sponsorship of several international utility companies.

The purpose of the multi-national effort was to develop a requirements document presenting a clear and complete statement of utility desires for design, construction and performance of a next generation of nuclear power plants. The main objectives for the establishment of the requirements document were to:

- establish a stabilized regulatory basis for future LWRs that would include the agreement of the US Nuclear Regulatory Commission (NRC) on resolution of outstanding licensing issues and severe accident issues, and which could provide a high assurance of licensability;
- provide a set of design requirements for a standardized plant which are reflected in individual reactor and plant supplier certification designs; and
- provide a set of technical requirements which are suitable for use in an ALWR investor bid package for eventual detailed design, licensing and construction, and which provide a basis for strong investor confidence that the risks associated with the initial investment to complete and operate the first ALWR are minimum.

This effort resulted in the EPRI advanced light water reactor (ALWR) Utility Requirements Document (URD) [8] that was the cornerstone of the US ALWR programme that addressed development of several LWR designs for the future.

The ALWR URD addresses the entire plant up to the grid interface, including nuclear steam supply system and balance of plant, and it applies to both BWRs and pressurized water reactors PWRs. It is organized in three volumes: Volume I summarizes ALWR programme policy statements and top tier requirements. Volumes II and III present the complete set of top tier and detailed requirements for specific ALWR design concepts. Volume II covers "evolutionary" ALWRs. These are simpler, much improved versions of existing LWRs, with a power output up to 1350 MW(e), employing conventional, but significantly improved, active safety systems. Volume III covers "passive" ALWRs, greatly simplified, smaller (i.e., of reference size 600 MW(e)) plants which employ primarily passive means (i.e. natural circulation, gravity drain, and stored energy) for essential safety functions.

The U.S. Nuclear Regulatory Commission (NRC) was directly involved in the process by reviewing the URD. In 1994 the NRC published a Safety Evaluation Report (SER) detailing their review of the requirements for each type of ALWR. Through the NRC review, the URD process contributed to improved stability in the regulatory basis for ALWRs by including agreements on outstanding licensing and severe accident issues.

The URD has been used in several ALWR design projects, specifically the design of the Combustion Engineering System 80+, the Westinghouse AP-600 and the General Electric ABWR. The ALWR design projects showed that the use of utility requirements such as those incorporated in the URD represents an essential element within the process of developing new LWR designs.

As noted above, the development of the EPRI URD was a multi-national effort; besides the US utilities there were significant contributions from interested utilities in Europe, — e.g. from Belgium, France, Germany, Italy, Spain, the Netherlands and the United Kingdom, — and in Asia, — from Japan, the Republic of Korea and Taiwan, China. Still, the EPRI URD strongly reflects the procedures, rules, regulations, codes and standards being used in the United States, and utilities in other countries, e.g. in the European Union, launched efforts to establish their own set of requirements taking into account the differences in procedures, rules, regulations, codes and standards, as well as other local conditions compared with those of the United States.

3.2.1.2. European Utility Requirements Document (EUR)

Since 1992 major European electricity producers have worked on a common set of European Utility Requirements (the EUR) for future LWR plants to establish specifications acceptable to the owners, the public and the authorities [9]. By this approach, nuclear plant designers can develop standard LWR designs that could be acceptable everywhere in Europe, and the utilities can open their consultations with vendors on common bases. Significant savings are expected in development and construction costs.

Development of the EUR was started by five partners in 1991, and now involves a group of ten organizations representing the major European utilities:

- British Energy plc / Nuclear Electric, United Kingdom,
- Desarrollo Tecnológico Nuclear (DTN), the Spanish organization for nuclear development
- Electricité de France
- Fortum and Teollisuuden Voima Oy, Finland
- SOGIN, Italy
- Tractebel, Belgium
- NRG, the Netherlands
- Vereinigung Deutscher Elektrizitätswerke (VDEW), the German utilities federation, Germany
- Vattenfall / FKA, Sweden
- Unterausschuss Kernenergie der Überlandwerke, Switzerland

Rosenergoatom for Russia became an associate member in 1998, and is in the process of becoming a full EUR member.

The EUR document is aimed at the LWR nuclear power plants to be built in Europe during the first decades of this century. The primary objective is to develop a common set of requirements that provide clear guidance to nuclear plant designers. It is a tool for harmonization of:

- the safety approaches, targets, criteria and assessment methods;
- design conditions and design methods;
- information required for assessment of safety, reliability and cost, and some related criteria;
- design requirements for the main systems and equipment; and
- equipment specifications and standards.

On this basis, standardized designs can be developed by the vendors and used by European electricity producers in an open market. The EUR provides European utilities and vendors a tool that allows them to develop, to assess and eventually to order modern LWR designs well fitted to the operator's needs. The main foreseen benefits are improved economic competitiveness and improvement in the licensing of new plants and in their public acceptance.

The EUR document is structured into four volumes:

• *Volume 1 Main policies and objectives*: this defines the major design objectives and presents the main policies that are implemented throughout the EUR document.

• *Volume 2 Generic nuclear island requirements*: this contains all the generic requirements and preferences of the EUR utilities for the nuclear island.

• *Volume 3 Application of EUR to specific designs*: this is divided into a number of subsets. Each subset is dedicated to a specific design that is of interest to the participating utilities. A subset includes a description of the design and an analysis of compliance vs. the generic requirements of Volume 2. It may also include design dependent requirements.

• *Volume 4 Power generation plant requirements*: this contains the generic requirements related to the power generation plant.

In parallel to the preparation of the EUR, the main LWR vendors have developed advanced designs for the European market, with reference to the EUR document. The EUR organization, in agreement with some of these vendors, has produced a Volume 3 subset to the EUR document that specifically addresses these new designs. Five subsets of Volume 3 dedicated to the BWR 90, the EPR, the EPP, the ABWR and SWR-1000 have been published. A subset on the WWER-1000 AES 92 design is foreseen.

3.2.1.3. Japanese Utility Requirements Document (JURD)

As noted above, a Japanese LWR standardization programme for improvement was initiated in the mid 1970s as a joint effort between the Government and the industry, under the leadership of the Ministry of International Trade and Industry (MITI), with the objective of improving and standardizing the LWR designs to enhance reliability and plant availability and to reduce occupational exposures. As a result, two advanced light water reactor designs, the advanced boiling water reactor (ABWR) and the advanced pressurized water reactor (APWR) have been developed. Two ABWR units have been built at Kashiwazaki-Kariwa, as units No. 6 &7; the first of these started commercial operation in November 1996, and the second in July 1997. The basic design for twin APWRs (Tsuruga 3 and 4) of the Japan Atomic Power Co (JAPCO) has been completed.

3.2.1.4. Korean Utility Requirements Document (KURD)

The purpose of the Korean Standard Requirements Document (KSRD) was to define the requirements for the series of PWRs built in the Republic of Korea in the mid-to-late 1990s, the Korean Standard Nuclear Power Plant Design (KSNP). The KSRD document is similar in scope to Volume II of the EPRI ALWR URD for PWRs, but contains some significant differences, with regard to the goal of establishing design and construction capability within the Republic of Korea. This document was completed in 1990.

Efforts in the Republic of Korea to develop user requirements for future plant designs began in 1993, as part of the next generation reactor development project. The objective of the development of this user requirements document, called the Korean Utility Requirements Document (KURD) was to delineate necessary features and characteristics of future reactors suitable to the Korean conditions, so that the direction of future reactor development in the Republic of Korea could be determined. The development of the user requirements has been being carried out in parallel with the basic design of the Korean Next Generation Reactor (KNGR), now named the Advanced Power Reactor 1400 (APR-1400) with the start of the Shin-kori 3,4 project in March 2001, in such a way that the requirements were established ahead of design work.

3.2.1.5. Chinese Utility Requirements for Evolutionary NPPs (CURD)

China is making the efforts to develop “Chinese Utility Requirements for Evolutionary NPPs”, which is divided into two phases:

- At first, “Chinese Utility Safety Requirements for Evolutionary NPPs” (CUSR) is prepared. The CUSR emphasizes particularly safety requirements.
- Secondly, CUSR will be complemented with by other requirements such as performance design requirements and then the comprehensive CURD will be established.

The preparation of *"Chinese Utility Safety Requirements for Evolutionary NPPs"*, was started in 2000 within an IAEA-TC project. IAEA provides technical support to the CUSR project with expert consultations, technical training and document reviews. The CUSR draft was finished in September 2002. IAEA provided three expert reviews of the CUSR draft at the end of 2002 covering all aspects of the CUSR.

The main objective of CUSR is to provide a comprehensive set of safety design requirements for Chinese Next Generation NPPs. The aims of the requirements are to promote:

- the safety approaches, targets, criteria and assessment methods;
- the design conditions;
- design objectives and criteria for the main systems and equipment;
- equipment specifications and standards;
- information required for the assessment of safety, reliability, and some of the corresponding criteria.

The safety requirements are also aimed at improving the safety and public acceptance of evolutionary NPPs in China:

- Setting safety targets for Chinese Next Generation NPPs;
- Promoting the ability of responses to safety problems;
- Setting low targets for accidents and routine radioactive release into the environment, and consideration of decommissioning aspects at design stage.

The CUSR documents are structured into 15 chapters, each chapter dealing with a specific topic. CUSR is applicable for Evolutionary PWR nuclear power plants with capacities from 600 MW up to 1500 MW.

The next phase of the CURD project will involve the relevant Chinese utilities. The CURD shall be presented as a complete statement of Chinese utility desires for design, construction and performance of the evolution of nuclear power plants in China. The main objectives are as follows:

- To establish a stabilized regulatory basis for future PWR NPPs. CURD shall satisfy the rules and regulations of the Chinese Government and National Nuclear Safety Administration of China on the outstanding licensing issues and severe accident issues. CURD also shall make contributions to improving public acceptance.

- To present a set of standard design conditions for the development of future Chinese NPPs, including the safety approaches, targets, criteria and assessment methods, performance targets, standard environmental conditions and standard design methods, and so on. The experiences of Chinese nuclear engineering practices shall be pooled in CURD.

- To provide a set of technical requirements that is suitable for bidding on design, licensing, construction, and so on.

3.2.2. Top Level Utility Requirements

3.2.2.1. General overview

Top-level requirements are summarized in Volume 1 of the EPRI ALWR requirements as well as in the European Utility Requirements. The utility requirements documents are addressed to designers and/or suppliers of LWR plants so that they can take the utility desires into consideration in the design at an early stage. They aim at promoting harmonization of:

- safety approaches, targets, criteria and assessment methods;
- standardization of design conditions;

- design objectives and criteria for the main systems and components;
- equipment specifications and standards; and
- information required for safety, reliability and cost assessment and some of the corresponding criteria.

The top-level requirements of Volume 1 of the European Utility Requirements are largely of a general nature, dealing with generic issues that are valid for all types of advanced LWRs. The EPRI URD, however, includes also some requirements for specific reactor types, namely for the passive reactors that are part of the US ALWR development programme.

The objective of the utility requirements, at least at the top tier level, is to prevent country specific conditions with respect to safety requirements, codes, standards, rules, regulations and laws leading to major differences in designs for different countries, or groups of countries. General top level requirements such as simplification, design margins, human factors, standardization, use of proven design, economic viability, etc. are quite similar in all countries, however, and should definitely not justify separate development programmes.

3.2.2.2. EPRI ALWR URD requirements

The first chapter of Volume I of the EPRI ALWR URD describes the US ALWR programme and objectives and scope of the requirements document.

Chapter 2 is entitled "ALWR program policy statements" and it delineates the requirements policies on a number of nuclear power plant issues to provide guidance for overall development of the URD, and to provide guidance to the plant designers in applying the requirements:

- pursue opportunities for **simplification** with very high priority;
- pursue **design margin** with high priority;
- take **human factors** into consideration making significant improvements in control room design;
- achieve **excellence in safety** for protection of the public, on-site personnel and the investment, placing emphasis in accident prevention as well as additional emphasis on mitigation, evaluating containment performance in severe accidents to assure adequate margin.
- include **safety design requirements** to meet NRC's regulations for the Licensing Design Basis with conservative, licensing based methods and **safety margin requirements** addressing investment protection and severe accidents on a best estimate basis;
- assure licensability/ **regulatory stabilization** through direct co-operation with the U.S. NRC ;
- form the foundation **standardized, certified** ALWR designs;
- employ **proven technology** and assure that a plant prototype is not required;
- design for **maintainability** to reduce operations and maintenance cost, reduce occupational exposure and facilitate repair and replacement of equipment;
- design to reduce the **construction schedule** relative to current plants, and complete engineering prior to initiation of construction;
- **quality assurance** leading to high quality design and construction work;
- **economics** to result in projected bus-bar costs with sufficient advantage over competing base-load electricity technologies to offset higher capital investment risk of nuclear projects;
- inherent **resistance to sabotage**; and
- design plant to be a **good neighbour** to its surrounding environment and population, by minimizing radioactive and chemical releases.

The top-tier ALWR design requirements of the URD are summarized in Table II from Ref. [10].

TABLE II. TOP-TIER ALWR DESIGN REQUIREMENTS OF THE EPRI URD

GENERAL UTILITY DESIGN REQUIREMENTS	
Unit size	applicable to a range of sizes up to 1350 MW(e) • Reference size for Evolutionary ALWR: 1200–1300 MW(e) • Reference size for Passive ALWR: 600 MW(e)
Safety system concept	• Evolutionary ALWR - simplified, improved active systems • Passive ALWR - passive systems; no safety-related ac power
design life	60 years
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 ≥ 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 (CDF)	Demonstrate by PRA that CDF is less than 10^{-5} per reactor year
• LOCA protection	No fuel damage for up to a 6-inch break
• Station blackout	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 with design pressure based on Licensing Design Basis pipe break
• Containment Margin	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.

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	
• Instrumentation and control 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 MW(e) evolutionary plant designed for 72 months or less 600 MW(e) 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

3.2.2.3. EUR requirements

Based on the Revision C of the EUR requirements some key requirements are summarized below:

Plant lifetime:

- The design life of the main pieces of plant equipment shall be at least 40 years without the need for refurbishment, and the design life of plant structures and non-replaceable components such as the reactor vessel and the primary containment shall be 60 years;
- structures and equipment that cannot meet these targets shall be replaceable.

Availability targets

- The target for overall availability is more than 90% on a reference period of 20 years for a 12 -month fuel cycle.
- The targets for outage duration are:
  - less than 20 days for normal outage with refuelling and regular maintenance
  - less than 30 days for outage with main turbine-generator overhaul
  - less than 40 days for outage with regulatory in-service inspection
  - less than 14 days for outage with refuelling only

- The targets for level of un-planned outage are:
  - Frequency of unplanned automatic scrams should be less than 1 per 7000 hrs critical
  - The annual forced unavailability factor should be less than 1.4% (less than 5 days per year).

Core and fuel:

- The core shall be capable of using up to 50% of standard fuel assemblies loaded with MOX (PuO_2-UO_2), together with standard UO_2 fuel assemblies;
- The core design shall show sufficient design margins that can be eventually used for operational purposes. To demonstrate that, the designer is requested to show that the core can produce its rated power under the following combination: (a) the most limitative of 50% of MOX fuel or 100% UO_2 fuel, (b) any refuelling interval between 12 and 24 months, (c) average discharge burnup below 55 MWd/kgHM for the UO_2 assemblies and 40MWd/kgHM for the MOX assemblies, and (d) load following and stretch-out as specified in the EUR document; and
- The fuel assembly mechanical design shall be capable of an assembly burnup of at least 60 MWd/kgHM for UO_2 fuel, and 45 MWd/kgHM for MOX fuel. These targets may be raised to 70 MWd/kgHM for UO_2 and 60 MWd/kgHM for MOX in the next 5 years.

Man-machine interface:

A strong emphasis should be placed on an operator friendly man-machine interface that

- is simple to understand especially for infrequently used systems;
- is oriented towards simplicity of operation and reduction of the risk of human error that could impair safety or performance. The number of different man-machine interfaces shall be restricted and the level of automation shall be such that the probability of human errors is minimised and that the consequences of such errors are limited. In the main control room, one single interface shall be used in all plant conditions, except when backup equipment is needed; and
- uses modern technology and ergonomic principles such as digital I&C, high-performance data transmissions busses and man machine interfaces based on workstations for control, backup and supervision.

Safety design requirements:

1. Accident resistance
   - Earthquakes: The standard plant shall be designed to withstand the effects of the Design Basis Earthquake (DBE), which is independent of site. For designing the standard plant, the free-field zero-period horizontal acceleration at ground level is set at 0.25g and is associated with three ground motion response spectra and three soil stiffness values that cover the majority of the potential sites in Western Europe.
   - Hazards from adjacent installations and transport activities shall be assessed site by site and dealt with if their probabilities of occurrence make them significant contributors to the PSA results.

With regard to an external explosion, the standard plant design shall incorporate provisions to withstand the effects of an external explosion which gives a pressure wave of 10^4 Pa and returning to zero in 300 ms.

2. Core damage prevention

- Core damage cumulative frequency shall be lower than 10^{-5} per reactor-year considering at power sequences, shutdown sequences, and hazards sequences.

3. Mitigation

- The containment system shall comprise a primary containment that is a leak-tight structure designed to withstand temperature and pressure conditions expected in design basis accidents and design extension conditions with appropriate design rules and a secondary containment designed to collect part or all of any releases from the primary containment.
- Severe accident risks shall be evaluated in a PSA and the cumulative frequency of exceeding the criteria for limited impact shall be lower than 10^{-6} per reactor-year. The EUR criteria for limited impact guarantee that the impact of any accident that meets such criteria is limited as follows: (a) no emergency action needed in the first 24 hours following the event beyond 800m, (b) no emergency action needed at any time beyond 3 km, (c) no long term emergency action needed beyond 800m, (d) restriction on consumption of foodstuff and crops limited to a reasonably small area for one or two years.
- If the containment atmosphere has not been inertized, the design shall assure that, assuming a hydrogen production equivalent to 100% of active fuel cladding / water interaction at a realistic rate, the average hydrogen concentration will not exceed 10% by volume in dry conditions, considering hydrogen control measures such as re-combiners and igniters.
- If the molten core cannot be proved to be coolable inside the reactor pressure vessel, preventing vessel failure, then specific provisions are needed to assure its coolability outside the vessel.

3.2.2.4. Korean Utility Requirements

The following summarizes top-tier requirements on which the APR-1400 design is based:

Economics and Performance:

Design life:	60 yrs
Availability goal:	greater than 90 percent
Construction period:	48 months for the Nth plant
Generation cost:	less than coal plant by 20 percent
Plant capacity:	1400 MW(e)
Daily load follow:	automatic
Refuelling interval:	18 months
I&C:	digital type

Safety:

Core melt frequency:	below 10^{-5} / RY
Seismic design:	0.3 g
Thermal margin:	more than 10 percent
ECCS:	4 trains with direct vessel injection

3.2.2.5. Chinese Utility Requirements Document for Evolutionary NPPs

The Chinese Utility Safety Requirements for Evolutionary NPPs (CUSR) are concerned with safety design requirements. The fundamental objective of CUSR meets the national safety rules and regulations of China and the IAEA requirements (such as INSAG series documents). CUSR also satisfies the safety objective and technology requirements presented in "Technology Policy of Several

Important Safety Issues for Design of Nuclear Power Plant To Be Built", which is established by NNSA in China recently.

The general safety requirements formulated in CUSR include:

A. Safety Design Requirements

1. Accident resistance
   - Design shall include sufficient reasonable margins;
   - The safe shutdown earthquake (SSE) shall be 0.25g;
   - Response time for operator to act after accidents/incidents shall be 30 minutes or more.

2. Core damage prevention
   - Core damage cumulative frequency shall be less than 10^{-5} per reactor year.

3. Mitigation
   - Severe accident analysis shall be evaluated in the combination of the deterministic method and PSA method.
   - Cumulative frequency of release from severe accidents shall be less than 10^{-6} per reactor year;
   - Significantly lower cumulative frequency for earlier or much larger releases.

Regarding the Chinese Utility Requirements, especially considering some specifications made by some Chinese utilities for bidding or/and the developing of next NPP projects in China, the CURD shall include the safety design requirements, the performance design requirements, the constructability requirements, the design process requirements and the economic requirements.

Besides the safety design requirements specified in CUSR, the other requirements are stated in the specifications by Chinese utilities.

B. Performance Design Requirements
- The plant shall be designed to operate for 60 years.
- The plant should be capable of operating on a fuel cycle with a refueling interval of 18 to 24 months.
- The plants shall be designed to achieve high availability during their operational lifetimes.
- With respect to release rates for normal operation and incidents, "utility limits and targets" are judged to be appropriate to take into account the national and international requirements and should be part of the objective of implementing the ALARA concept.

3.3. Example of Application of Utility Requirements [Finland]

As has been described in Section 1.2, in 2002 Teollisuuden Voima Oy (TVO), Finland, invited bids from NPP suppliers for a new unit. In preparation for this, the official document of bid inquiry specifications (BIS) that laid out the requirements for the plant and other conditions related to the bid was prepared. As is discussed in Ref. [11], this compilation of a complete set of utility requirements for a new plant is a very large task that was facilitated by the comprehensive and detailed model of the EUR document. TVO has actively participated in the EUR since the mid-1990s, and the EUR document is well known to the staff. This contributed to the efficiency of the preparation of the bid inquiry specifications, as described below, based on information in Ref. [11].

By definition, the EUR is a reference for the technical specifications of a NPP bid inquiry. Comparison between the EUR and the Finnish NPP licensing requirements, the YVL guides, had been performed in 1997 and results were considered by the EUR organization in compiling Revision C of Volumes 1 and 2 (see **Section 3.2.1.2**).

A bid specification had previously been developed in 1991 in Finland by Perusvoima Oy (PEVO), a joint venture owned by Imatran Voima Oy (IVO) and TVO[14]. The PEVO documentation had been based on the experiences gained in the Loviisa and Olkiluoto projects and therefore took account of the local conditions valid in Finland at that time. Both the EUR and the PEVO documentation were used in preparing the BIS for the bids that TVO invited in 2002. Technical requirements were specified by using the EUR document as a reference, while other parts were either based on PEVO material or completely rewritten.

The BIS is compiled so that they provide a basis for the contract documents while being an outline of the bid. The BIS has the following parts:

- Instructions for Bidders;
- Terms and Conditions;
- Scope of Supply and Services;
- Project Implementation;
- General Technical Requirements;
- Power Generation Plant Requirements; and
- Nuclear Fuel.

The Instructions for Bidders specify in detail how the bid is to be submitted and the terms and conditions define the basis of the contract. The Scope of Supply and Services addresses the division of responsibilities between the supplier and the purchaser, and are generally based on Ref. [12]. The Project Implementation defines different alternatives for executing the project in accordance with objectives for schedule, cost, etc. The General Technical Requirements relate to EUR Volumes 1 and 2, and the Power Generation Requirements relate to EUR Volume 4.

While the EUR defines requirements for a European standard NPP, the BIS for the Finnish NPP adapts requirements to the local conditions, the Finnish licensing requirements, and the operating experience of the Finnish NPPs. Importantly, the detailed model of the EUR greatly facilitated the work of preparing the utility requirements for the new NPP in Finland within a reasonable time period.

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CHAPTER 4. LARGE SIZE ADVANCED LWR DESIGNS (700 MW(e) AND LARGER)

4.1 ABWR (GENERAL ELECTRIC, USA / HITACHI LTD. AND TOSHIBA CORP., JAPAN)

The cumulative experience of 40 years of evolutionary design, development and operating experience of BWRs around the world has led to significant enhancements in safety, operation and maintenance (O&M) practices, economics, radiation exposure and radwaste reduction. The development of the ABWR started in 1978 as an international co-operation between five BWR vendors: GE of the USA, Hitachi and Toshiba of Japan, and European BWR vendors. An advanced engineering team (AET), that comprised personnel from all five companies, developed a conceptual design of an improved BWR, with the support of the different company's home offices.

This conceptual design was received favourably by Tokyo Electric Power Company (TEPCO) and other Japanese utilities, and as a result, the ABWR was included in the third standardization programme of Japan from 1981. Preliminary design and numerous development and verification tests were carried out simultaneously by Toshiba, Hitachi and GE together with six Japanese utilities and the Japanese government toward realization of the plant. From 1987 GE, Hitachi and Toshiba started project engineering, detailed design and preparation of licensing documents for the Kashiwazaki-Kariwa nuclear power station units 6&7, which were then ordered by TEPCO from this international consortium. These two units were taken into commercial operation in 1996 and 1997 respectively.

The following design description is based on the US version of the ABWR design that received design certification from the US NRC in May 1997, and it also reflects subsequent improvements made in Japan after the former report "IAEA-TECDOC-968" [1].

4.1.1 Introduction

The design of the advanced boiling water reactor (ABWR) represents a complete design for a nominal 1300 MWe power plant. The inclusion of such features as reactor internal pumps, fine motion control rod drives, multiplexed digital fiber-optic control systems, and an advanced control room are examples of the type of advancements over previous designs that have been incorporated to meet the ABWR objectives.

The ABWR design objectives include: 60 year plant life from full power operating license date, 87% or greater plant availability, less than one unplanned scram per year, 24 month refuelling interval, personnel radiation exposure limit of 100 man-rem/year, core damage frequency of less than 10^{-5}/ reactor year, limiting significant release frequency to 10^{-6}/reactor year, and reduced radwaste generation.

The principal design criteria governing the ABWR standard plant encompass two basic categories of requirements: those related to either a power generation function or a safety related function.

General power generation design criteria

The plant is designed to produce electricity from a turbine generator unit using steam generated in the reactor.

Heat removal systems are designed with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and abnormal operational transients. Backup heat removal systems are designed to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage.

The fuel cladding, in conjunction with other plant systems, is designed to retain its integrity so that the consequences of any equipment failures are within acceptable limits throughout the range of normal operational conditions and abnormal operational transients for the design life of the fuel.

Control equipment is designed to allow the reactor to respond automatically to load changes and abnormal operational transients. Reactor power level is manually controllable.

Interlocks or other automatic equipment are designed as backup to procedural control to avoid conditions requiring the functioning of safety related systems or engineered safety features.

General safety design criteria

The plant is designed, fabricated, erected and operated in such a way that the release of radioactive material to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations, for abnormal transients and for accidents.

The reactor core is designed so that its nuclear characteristics counteract a power transient. The reactor is designed so that there is no tendency for divergent oscillation of any operating characteristics considering the interaction of the reactor with other appropriate plant systems.

Safety related systems and engineered safety features function to ensure that no damage to the reactor coolant pressure boundary results from internal pressures caused by abnormal operational transients and accidents. Where positive, precise action is immediately required in response to abnormal operational transients and accidents, such action is automatic and requires no decision or manipulation of controls by plant operations personnel.

The design of safety related systems, components and structures includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the plant site.

Standby electrical power sources have sufficient capacity to power all safety-related systems requiring electrical power concurrently. Standby electrical power sources are designed to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

A containment is provided that completely encloses the reactor systems, drywell, and pressure suppression “wetwell” chambers. The containment employs the pressure suppression concept.

A safety envelope is provided that basically encloses the containment, with the exception of the areas above the containment top slab and drywell head. The containment and safety envelope, in conjunction with other safety related features, limit radiological effects of design basis accidents to less than the prescribed acceptable limits. The reactor building surrounds the containment/safety envelope and serves as a secondary containment.

Provisions are made for removing energy from the containment as necessary to maintain the integrity of the containment system following accidents that release energy to the containment.

Emergency core cooling is designed to limit fuel cladding temperature to less than the limits of 10CFR50.46 (2200°F or 1204°C) in the event of a design basis loss of coolant accident (LOCA). The emergency core cooling is designed for continuity of core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary piping. Emergency core cooling is initiated automatically when required regardless of the availability of off site power supplies and the normal generating system of the plant.

The control room is shielded against radiation so that continued occupancy under design basis accident conditions is possible. In the event that the control room becomes uninhabitable, it is possible to bring the reactor from power range operation to cold shutdown conditions by utilizing alternative controls and equipment that are available outside the control room.

Fuel handling and storage facilities are designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel as necessary to meet operating and off-site dose constraints.

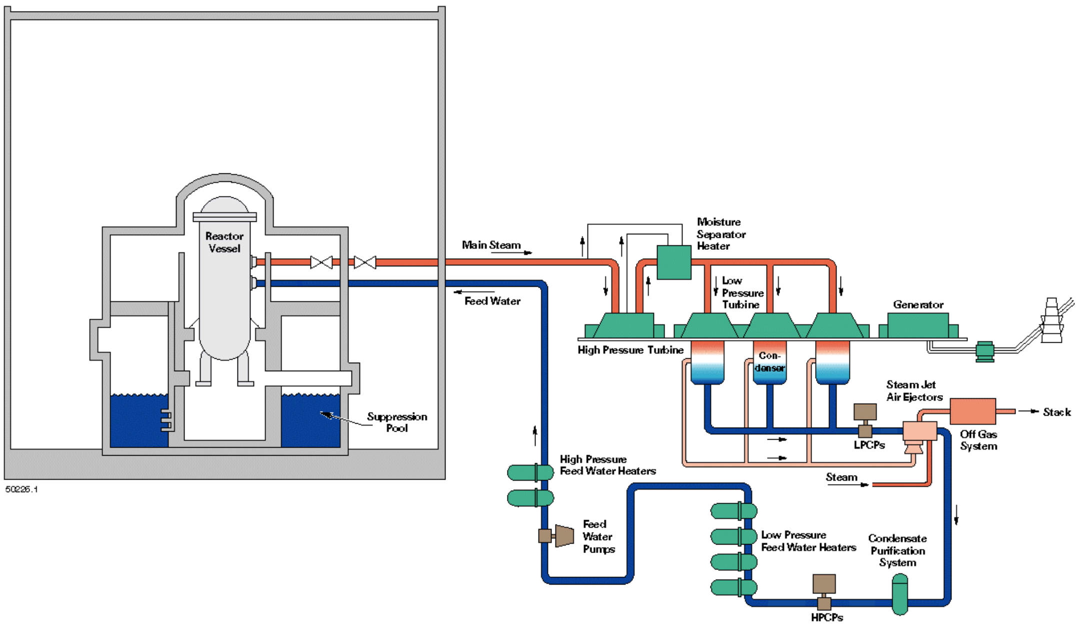

FIG. 4.1-1. ABWR - steam cycle.

4.1.2 Description of the nuclear systems

4.1.2.1 Primary circuit and its main characteristics

Figure 4.1-1 illustrates the ABWR steam cycle. The primary functions of the nuclear boiler system are:

(1) to deliver steam from the reactor pressure vessel (RPV) to the turbine main steam system,
(2) to deliver feedwater from the condensate and feedwater system to the RPV,
(3) to provide overpressure protection of the reactor coolant pressure boundary,
(4) to provide automatic depressurization of the RPV in the event of a loss of coolant accident (LOCA) where the RPV does not depressurize rapidly, and
(5) with the exception of monitoring the neutron flux, to provide the instrumentation necessary for monitoring conditions in the RPV such as RPV pressure, metal temperature, and water level instrumentation.

The main steam lines (MSLs) are designed to direct steam from the RPV to the main steam system of the turbine, and the feedwater lines (FWLs) to direct feedwater from the condensate and feedwater system to the RPV.

The main steam line flow limiter, a flow restricting venturi built into the RPV MSL nozzle of each of the four main steam lines, limits the coolant blowdown rate from the reactor vessel to a (choke) flow rate equal to or less than 200% of rated steam flow at 7.07 MPa (1025 psig) upstream gauge pressure in the event a main steam-line break occurs anywhere downstream of the nozzle.

There are two main steam isolation valves (MSIVs) welded into each of the four MSLs, one inner MSIV in the containment and one outer MSIV outside the containment. The MSIVs are Y-pattern globe valves. The Y-pattern configuration permits the inlet and outlet flow passages to be streamlined to minimize pressure drop during normal steam flow.

The nuclear pressure relief system consists of safety/relief valves (SRVs) located on the main steam lines (MSLs) between the RPV and the inboard main steam line isolation valve. There are 18 SRVs distributed on the four MSLs. The SRVs are designed to provide three main protection functions: overpressure safety, overpressure relief, and depressurization operation, which is discussed below separately.

The automatic depressurization subsystem (ADS) consists of the eight SRVs and their associated instrumentation and controls. The ADS designated valves open automatically for events involved with small breaks in the nuclear system process barrier or manually in the power actuated mode when required. The ADS designated valves are capable of operating from either ADS LOCA logic or overpressure relief logic signals. The ADS accumulator capacity is designed to open the SRV against the design drywell pressure following failure of the pneumatic supply to the accumulator.

4.1.2.2 Reactor core and fuel design

The ABWR core configuration consists of 872 bundles. The rated core power is 3926 MWt, which corresponds to a 50.6 kW/l power density. The lower power density results in improved fuel cycle costs and greater maneuverability. Since the ABWR utilizes reactor internal pumps (RIPs) to control the recirculation flow through the core, the reactivity control is maintained by a combination of changes in core flow, control rod position and by the inclusion of burnable poison in the fuel.

Control rod drive system

The control rod drive (CRD) system is composed of three major elements: the fine motion control rod drive (FMCRD) mechanisms; the hydraulic control unit (HCU) assemblies, and the control rod drive hydraulic (CRDH) subsystem.

The FMCRDs (Figure 4.1-2 which shows a cross-section of a FMCRD) are designed to provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) in response to manual or automatic signals from the reactor protection system (RPS). In addition to hydraulic-powered scram, the FMCRDs also provide electric-motor-driven run-in of all control rods as a path to rod insertion that is diverse from the hydraulic powered scram. The hydraulic power required for scram is provided by high pressure water stored in the individual HCUs. The HCUs also provide the flow path for purge water to the associated drives during normal operation. The CRDH subsystem supplies high pressure demineralized water which is regulated and distributed to provide charging of the HCU scram accumulators, purge water flow to the FMCRDs, and backup makeup water to the RPV when the feedwater flow is not available.

There are 205 FMCRDs mounted in housings welded into the RPV bottom head. Each FMCRD has a movable hollow piston tube that is coupled at its upper end, inside the reactor vessel, to the bottom of a control rod. The piston is designed such that it can be moved up or down, both in fine increments and continuously over its entire range, by a ball nut and ball screw driven at a nominal speed of 30 mm/s by the electric stepper motor.

In response to a scram signal, the piston rapidly inserts the control rod into the core hydraulically using stored energy in the HCU scram accumulator. The FMCRD design includes an electro-mechanical brake on the motor drive shaft and a ball check valve at the point of connection with the scram inlet line. These features prevent control rod ejection in the event of a failure of the scram insert line. There are 103 HCUs, each of which provides sufficient volume of water stored at high pressure in a pre-charged accumulator to scram two FMCRDs at any reactor pressure.

In Japan, a sealless FMCRD, a new type of FMCRD, has been developed and deployed in Hamaoka Unit-5, currently under construction. Its basic design is shown in Figure 4.1-2, on the right. The purpose of the sealless FMCRD is to eliminate FMCRD shaft penetration through the pressure boundary, utilizing a magnet coupling to transmit drive force from the electric motor. This new system eliminates the ground packing and leak detection system of the conventional FMCRD, and enhances reliability and maintainability. In addition, the

FMCRD's step width requirement has been mitigated by improvements of fuel design, and a change in the motor-drive system from a stepper motor and inverter power source to an induction motor with AC power source and contactor.

4.1.2.3 Fuel handling and transfer systems

The reactor building is supplied with a refuelling machine for fuel movement and servicing plus an auxiliary platform for servicing operations from the vessel flange level.

The refuelling machine is a gantry crane, which spans the reactor vessel and the storage pools on bedded tracks in the refuelling floor. A telescoping mast and grapple suspended from a trolley system is used to lift and orient fuel bundles for placement in the core and/or storage racks. Two auxiliary hoists, one main and one auxiliary monorail trolley-mounted, are provided for in-core servicing. Control of the machine is from an operator station on the refuelling floor.

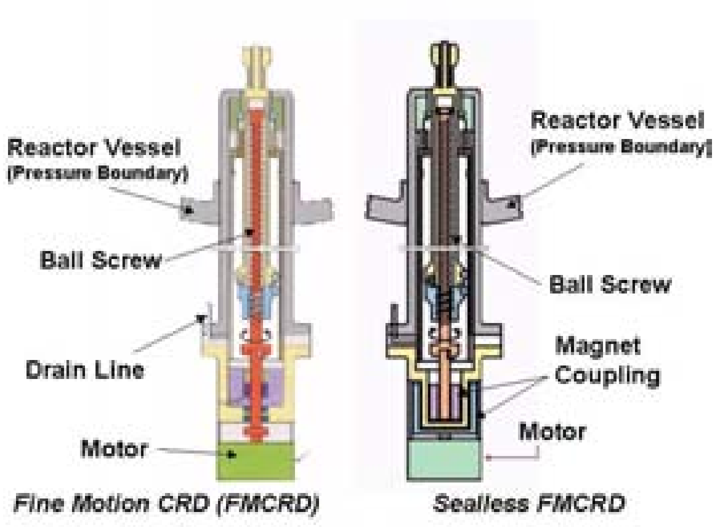

FIG. 4.1-2. Cross-section of fine motion control rod drive (FMCRD) and seallessFMCRD

A position indicating system and travel limit computer are provided to locate the grapple over the vessel core and prevent collision with pool obstacles. The mast grapple has a redundant load path so that no single component failure results in a fuel bundle drop. Interlocks on the machine: (1) prevent hoisting a fuel bundle over the vessel unless an all-control-rod-in permissive is present; (2) limit vertical travel of the fuel grapple to provide shielding over the grappled fuel during transit; (3) prevent lifting of fuel without grapple hook engagement and load engagement.

Storage racks are provided for the temporary and long-term storage of new and spent fuel and associated equipment. The new and spent fuel storage racks use the same configuration and prevent inadvertent criticality.

Racks provide storage for spent fuel in the spent fuel storage pool in the reactor building. New fuel, 40% of the reactor core, is stored in the new fuel storage vault in the reactor building. The racks are top loading, with fuel bail extended above the rack. The spent fuel racks have a minimum storage capacity of 270% of the reactor core, which is equivalent to a minimum of 2354 fuel storage positions. The new and spent fuel racks maintain a subcriticality of at least 5% Δk under dry or flooded conditions. The rack arrangement prevents accidental insertion of fuel assemblies between adjacent racks and allows flow to prevent the water from exceeding 100°C.

4.1.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel (RPV) system consists of (1) the RPV and its appurtenances, supports and insulation, excluding the loose parts monitoring system, and (2) the reactor internal components enclosed by the vessel, excluding the core (fuel assemblies, control rods, in-core nuclear instrumentation and neutron sources), reactor internal pumps (RIPs), and control rod drives (CRDs). The RPV system is located in the primary containment.

The reactor coolant pressure boundary (RCPB) portion of the RPV and its appurtenances act as a radioactive material barrier during plant operation.

Certain reactor internals support the core, flood the core during a loss of coolant accident (LOCA) and support safety related instrumentation. Other RPV internals direct coolant flow, separate steam, hold material surveillance specimens, and support instrumentation utilized for plant operation.

The RPV system provides guidance and support for the CRDs. It also distributes sodium pentaborate solution when injected from the standby liquid control (SLC) system.

The RPV system restrains the CRD to prevent ejection of the control rod connected with the CRD in the event of a failure of the RCPB associated with the CRD housing weld. A restraint system is also provided for each RIP in order to prevent the RIP from becoming a missile in the event of a failure of the RCPB associated with the RIP casing weld.

The RPV is a vertical, cylindrical vessel of welded construction with removable top head and head closure bolting seals. Through the use of large forged rings, the number of welds in the RPV is reduced. The main body of the installed RPV has a cylindrical shell, flange, bottom head, RIP casings, penetrations, brackets, nozzles, and the shroud support, which has a pump deck forming the partition between the RIP suction and discharge. The shroud support is an assembly consisting of a short vertical cylindrical shell, a horizontal annular pump deck plate and vertical support legs.

An integral reactor vessel support skirt supports the vessel on the reactor pressure vessel pedestal. Anchor bolts extend from the pedestal through the flange of the skirt. RPV stabilizers are provided in the upper portion of the RPV to resist horizontal loads. Lateral supports for the CRD housings and in-core housings are provided.

The large RPV volume provides a large reserve of water above the core, which translates directly into a much longer period of time (compared to prior BWRs) before core uncovery is likely to occur as a result of feedwater flow interruption or a LOCA. This gives an extended period of time during which automatic systems or plant operators can re-establish reactor inventory control using any of several normal, non-safety-related systems capable of injecting water into the reactor. Timely initiation of these systems precludes the need for activation of emergency safety equipment. The large RPV volume also reduces the reactor pressurization rates that develop when the reactor is suddenly isolated from the normal heat sink which eventually leads to actuation of the safety-relief valves.

Reactor internals

The ABWR RPV and internals are illustrated in Figure 4.1-3. The major reactor internal components in the RPV System are: (1) Core support structures, and (2) Other reactor internals.

The Core support structures encompass: the shroud, shroud support and a portion of CRD housings inside the reactor internals RPV, core plate, top guide, fuel supports, and control rod guide tubes (CRGTs).

Other reactor internals are:

- Feedwater spargers, shutdown cooling (SDC) and low pressure core flooder (LPFL) spargers for the Residual heat removal (RHR) system, high pressure core flooder (HPCF) spargers and couplings, and a portion of the in-core housings inside the RPV and in-core guide tubes (ICGTs) with stabilizers.

- Surveillance specimen holders, shroud head and steam separators assembly and the steam dryer assembly.

Reactor recirculation pumps

The reactor recirculation system (RRS) features an arrangement of ten variable speed reactor coolant recirculation pumps. The pumps with motors are mounted in the bottom of the RPV, and are thus termed reactor internal pumps (RIPs). The RIPs provide forced circulation of the reactor coolant through the lower plenum of the reactor and up through the lower grid, the reactor core, steam separators, and back down the downcomer annulus.

The recirculation flow rate is variable over a "flow control range," from minimum flow established by certain pump performance characteristics to above maximum flow required to obtain rated reactor power.

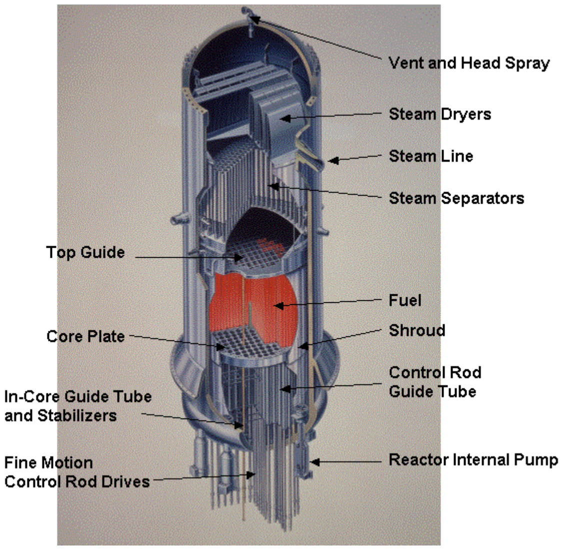

FIG. 4.1-3. ABWR – reactor pressure vessel and internals

By regulating the flow rate, the reactor power output can be regulated over an approximate range from 70 to 100% of rated output, without moving control rods. RIP performance is adequate to allow plant operation at 100% power with only 9 of the 10 pumps in operation.

Each RIP includes a device which prevents reverse RIP motor rotation by reverse flow induced torque. The RIP motor cooling is provided by an auxiliary impeller mounted on the bottom of the motor rotor, which circulates water through the RIP motor and its cooling heat exchanger. The heat exchangers are cooled by the reactor building cooling water system. Figure 4.1-4 illustrates a cross-section of a RIP.

4.1.2.5 Reactor auxiliary systems

The main auxiliary systems in the ABWR nuclear island consist of the reactor building cooling water (RBCW) system, the reactor water cleanup (RWCU) system, the fuel pool cooling and cleanup (FPCU) system and the suppression pool cleanup (SPCU) system. In addition there are many other auxiliary systems such as instrument and service air, condensate and demineralized water transfer, chilled water, HVAC, equipment drain, floor drain and other systems which are basically the same as on past BWR plants and are not covered in this report since the designs are all well proven.

The RBCW system consists of piping, valves, pumps and heat exchangers which are used to provide cooling water to the various consumers in the nuclear island. The system is divided into three separate safety divisions, each with its own pump and heat exchanger, to provide cooling water to equipment in the three ECCS and RHR safety divisions. The RBCW system also provides cooling water to equipment in non safety systems such as the RWCU, FPCU and other systems and equipment that require cooling water.

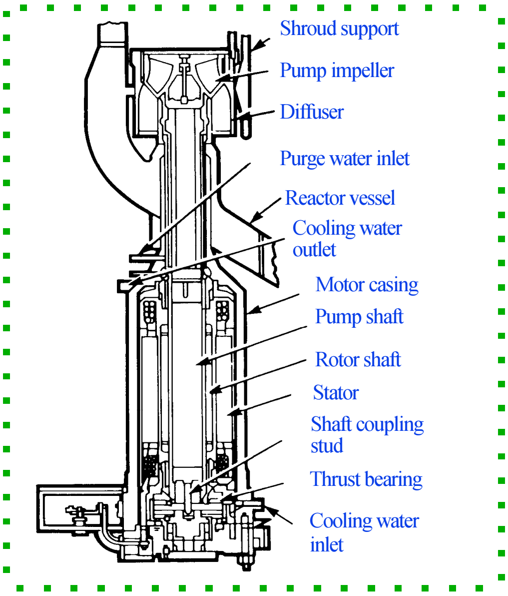

FIG. 4.1-4. Cross-section of reactor internal pump

The RWCU heat exchangers are cooled by water from the plant service water or ultimate heat sink depending on unique site conditions.

The RWCU system consists of piping, valves, pumps, heat exchangers and filter demineralizers which are used to remove impurities from the reactor primary coolant water to maintain water quality within acceptable limits during the various plant operating modes. The RWCU design for ABWR is basically the same as on previous BWRs with the following exceptions: 1) the RWCU pumps are located downstream of the regenerative and non-regenerative heat exchangers to reduce the pump operating temperature and improve pump seal and bearing performance, and 2) two 1% capacity systems are used instead of only one 1% system, as found in previous BWRs.

The FPCU and SPCU systems consist of piping, valves, pumps, heat exchangers and filter-demineralizers which are used to remove decay heat from the spent fuel storage pool and to remove impurities from the water in the spent fuel pool and dryer/separator pool and suppression pool to maintain water quality within acceptable limits during various plant operating modes. The filter demineralizer in the FPCU system is shared by the SPCU system for cleaning the suppression pool water. The FPCU and SPCU systems are basically the same as on previous BWRs.

4.1.2.6 Operating characteristics

The ABWR design incorporates extensive automation of the operator actions which are required during a normal plant startup, shutdown and power range manoeuvers. The automation features adopted for the ABWR are designed for enhanced operability and improved capacity factor, relative to conventional BWR designs. However, the extent of automation implemented in the ABWR has been carefully selected to ensure that the primary control of plant operations remains with the operators. The operators remain fully cognizant of the plant status and can intervene in the operation at any time, if necessary.

The ABWR control room design provides the capability for a single operator to perform all required control and monitoring functions during normal plant operations as well as under emergency plant conditions. One man operation is possible due to implementation of several key design features: the wide display panel for overall plant monitoring, plant-level automation, system-level automation via sequence master control switches, the compact main control console design, and implementation of operator guidance functions which display appropriate operating sequences on the main control panel CRTs. The role of the operator will primarily be one of monitoring the status of individual systems and the overall plant and the progress of automation sequences, rather than the traditional role of monitoring and controlling individual system equipment. However, to foster a team approach in plant operation and to maintain operator vigilance, the operating staff organization for the reference ABWR control room design is based upon having two operators normally stationed at the control console.

The incorporation of Reactor internal pumps (RIPs) allows power changes of up to 30% of rated power to be accomplished automatically by recirculation flow control alone, thus providing automatic electrical load-following capability for the ABWR without the need to adjust control rod settings.

The ABWR fine-motion control rod drives (FMCRDs) are moved electronically in small increments during normal operation, allowing precise power management. The FMCRDs are inserted into the core hydraulically during emergency shutdown, with the backup provision for continuous electronic insertion.

4.1.3 Description of turbine generator plant systems

4.1.3.1 Turbine generator plant

The main turbine

The main turbine is a six flow, tandem compound, single reheat, 1800 rpm machine with 1320.8 mm (52 in.) last stage blades. The turbine has one duel-exhaust high pressure section and three dual-exhaust low pressure sections. The cycle uses conventional moisture separator reheaters with single stage reheat for the cross-around steam.

Extraction steam from the high and low-pressure turbine extraction nozzles is conveyed to the high and low-pressure feedwater heaters, respectively. The feedwater heating systems are designed to provide a final feedwater temperature of 216°C (420°F) at 100 percent nuclear boiling rate. This cycle yields a gross generator output of approximately 1 385 000 kW with a thermal reactor output of 3 926 000 kW.

Turbine bypass system

The turbine bypass system (TBP) provides the capability to discharge main steam from the reactor directly to the condenser to minimize step load reduction transient effects on the reactor coolant system. The TBP is also used to discharge main steam during reactor hot standby and cooldown operations.

The TBP consists of a three-valve chest that is connected to the main steam lines upstream of the turbine stop valves, and of three dump lines that separately connect each bypass valve outlet to one condenser shell. The system is designed to bypass at least 33% of the rated main steam flow directly to the condenser. The TBP, in combination with the reactor systems, provides the capability to shed 40% of the turbine generator rated load without reactor trip and without the operation of safety/relief valves. A load rejection in excess of 40% is expected to result in reactor trip but without operation of any steam safety valve.

The turbine bypass valves are opened by redundant signals received from the Steam bypass and pressure control system whenever the actual steam pressure exceeds the preset steam pressure by a small margin. This occurs when the amount of steam generated by the reactor cannot be entirely used by the turbine. This bypass demand signal causes fluid pressure to be applied to the operating cylinder, which opens the first of the individual valves. As the bypass demand increases, additional bypass valves are opened, dumping the steam to the condenser. The bypass valves are equipped with fast acting servo valves to allow rapid opening of bypass valves upon turbine trip or generator load rejection.

Main condenser

The main condenser, which does not serve or support any safety function and has no safety design basis, is a multipressure three-shell type deaerating type condenser. During plant operation, steam expanding through the low pressure turbines is directed downward into the main condenser and condensed. The main condenser also serves as a heat sink for the turbine bypass system, emergency and high level feedwater heater and drain tank dumps, and various other startup drains and relief valve discharges.

Each condenser shell has two tube bundles. Circulating water flows in series through the three shells. The Condenser circulating water system (CCW) is designed to permit any portion of the condenser to be isolated and removed from service.

The main condenser is located in the turbine building in pits below the operating floor and is supported by the turbine building base mat. The Condensate return tank is located in the turbine building above its connection to the low pressure condenser shell.

Since the main condenser operates at a vacuum, radioactive leakage to the atmosphere cannot occur. Circulating water leakage into the shell side of the main condenser is detected by measuring the conductivity of the condensate. Conductivity of the condensate is continuously monitored at selected locations in the condenser. Leak detection trays are included at all tube-to-tubesheet interfaces. Provisions for early leak detection are provided at tubesheet trays and in each hotwell section. The hotwell is divided into sections to allow for leak detection and location. Conductivity and sodium content are alarmed in the main control room and preclude any automatic bypass of the demineralizers.

The main condenser evacuation system (MCES) removes the non-condensable gases from the power cycle. The MCES removes the hydrogen and oxygen produced by radiolysis of water in the reactor, and other power cycle non-condensable gases, and exhausts them to the offgas system during plant power operation, and to the turbine building compartment exhaust system at the beginning of each startup.

The MCES consists of two 100%-capacity, double stage, steam jet air ejector (SJAE) units (complete with intercondenser) for power plant operation where one SJAE unit is normally in operation and the other is on standby, as well as a mechanical vacuum pump for use during startup. The last stage of the SJAE is a non-condensing stage.

During the initial phase of startup, when the desired rate of air and gas removal exceeds the capacity of the steam jet air ejectors, and nuclear steam pressure is not adequate to operate the SJAE units, the mechanical vacuum pump establishes a vacuum in the main condenser and other parts of the power cycle. The discharge from the vacuum pump is then routed to the turbine building compartment exhaust system, since there is then little or no effluent radioactivity present. Radiation detectors in the turbine building compartment exhaust system and plant vent alarm in the main control room if abnormal radioactivity is detected. Radiation monitors are provided on the main steamlines which trip the vacuum pump if abnormal radioactivity is detected in the steam being supplied to the condenser.

The SJAEs are placed in service to remove the gases from the main condenser after a pressure of about 34 to 51 kPa absolute is established in the main condenser by the mechanical vacuum pump and when sufficient nuclear steam pressure is available.

During normal power operations, the SJAEs are normally driven by cross-around steam, with the main steam supply on automatic standby. The main steam supply, however, is normally used during startup and low load operation, and auxiliary steam is available for normal use of the SJAEs during early startup, should the mechanical vacuum pump prove to be unavailable.

4.1.3.2 Condensate and feedwater systems

The condensate and feedwater system are designed to provide a dependable supply of high-quality feedwater to the reactor at the required flow, pressure, and temperature. The condensate pumps take the deaerated condensate from the main condenser hotwell and deliver it through the steam jet air ejector condenser, the gland steam condenser, the off-gas condenser, the condensate demineralizer, and through three parallel strings of four low pressure feedwater heaters to the reactor feed pumps' section. The two reactor feed pumps each have an approximate capacity of 4 600 m^3/h. They each discharge through two stages of high pressure heaters (two parallel strings) to the reactor. Each reactor feedwater pump is driven by an adjustable speed synchronous motor. The drains from the high pressure heaters are pumped backward to the suction of the feed pumps.

Two 22 in. (559 mm) feedwater lines transport feedwater from the feedwater pipes in the steam tunnel through RCCV penetrations to horizontal headers in the upper drywell which have three 12 in. (305 mm) riser lines that connect to nozzles on the RPV. Isolation check valves are installed upstream and downstream of the RCCV penetrations and manual maintenance gate valve are installed in the 22-in. lines upstream of the horizontal headers.

4.1.3.3 Auxiliary systems

The turbine building cooling water system (TBCW), which is a non safety related system, removes heat from the auxiliary equipment in the turbine building and rejects this heat to the turbine building service water (TBSW) system. The TBSW system rejects the heat taken from the TBCW system to the power cycle heat sink which is part of the Circulating water system.

The service air (SAIR) system provides compressed air for general plant use. The SAIR system also provides backup to the instrument air (IAIR) system in the event that the IAIR system pressure is lost. The IAIR system provides compressed air for pneumatic equipment, valves, controls and instrumentation outside the primary containment.

4.1.4 Instrumentation and control systems

4.1.4.1 Design concepts, including control room

The ABWR control and instrument systems are designed to provide manual and automatic means to control plant operations and initiate protective actions should plant upset conditions occur. The ABWR utilizes digital controllers, interfacing with plant equipment, sensors and operator controls through a multiplexing system for signal transmission to achieve these functions.

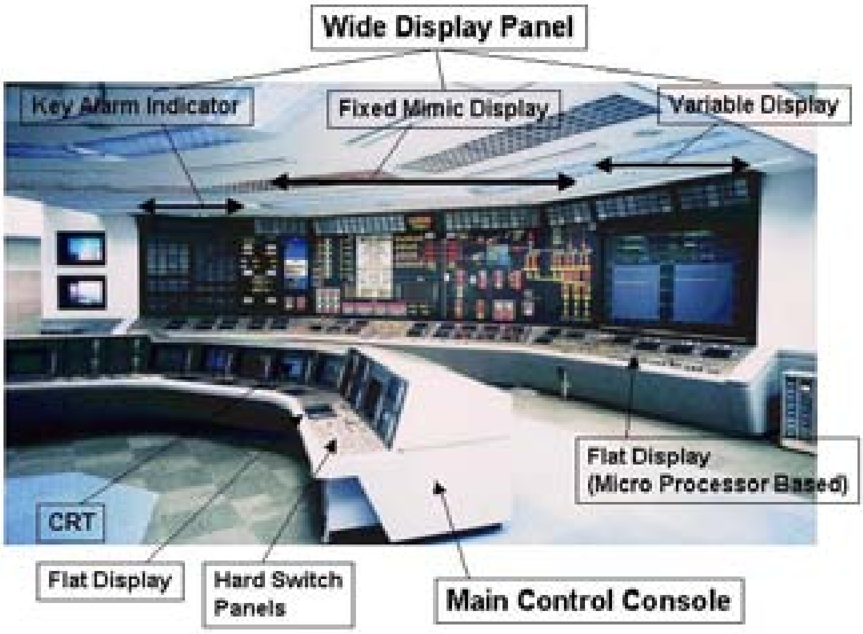

FIG. 4.1-5. ABWR – main control room (KK-6).

The key distinguishing simplification features for plant control and monitoring include:

- Enhanced man-machine interface design
- Automated plant operations
- Simplified neutron monitoring system
- Reduction in number of nuclear boiler instruments
- Fault-tolerant safety system logic and control
- Standardized digital control and measurement
- Multiplexing of plant control signals.

Multiplexed signal transmission using high speed fiber optic data links is combined with digital technology to integrate control and data acquisition for both reactor and turbine plants. Multiplexing significantly reduces the quantities of control cables which need to be installed during construction, thereby reducing the construction cost, and facilitates automation of plant operations.

Performance monitoring and control, and power generator control subsystem functions are provided by the Process computer system to support efficient plant operation and automation.

The main control room panels (MCRPs) consist of an integrated set of operator interface panels (e.g., main control console, large display panel), as depicted in Figure 4.1-5. The safety related panels are seismically qualified and provide grounding, electrical independence and physical separation between safety divisions and non-safety-related components and wiring.

The MCRPs and other main control room operator interfaces are designed to provide the operator with information and controls needed to safely operate the plant in all operating modes, including startup, refuelling, safe shutdown, and maintaining the plant in a safe shutdown condition. Human factors engineering principles have been incorporated into all aspects of the ABWR main control room design.

The liquid and solid radwaste systems are operated from control panels in the radwaste control room. Programmable controllers are used in this application.

4.1.4.2 Reactor protection and other safety systems

The safety system logic and control (SSLC) provides a centralized facility of implementing safety related logic functions. The SSLC is configured as a four-division data acquisition and control system, with each division containing an independent set of microprocessor-based software controlled logic processors.

The reactor protection system (RPS) is an overall complex of instrument channels, trip logic, trip actuators, manual controls, and scram logic circuitry that initiates the rapid insertion of control rods by hydraulic force to scram the reactor when unsafe conditions are detected. The RPS uses the functions of the essential multiplexing subsystem (EMS) and the SSLC system to perform its functions.

The remote shutdown system (RSD) is designed to safety shut down the reactor from outside the main control room. The RSD provides remote manual control to the systems necessary to: (a) achieve prompt hot shutdown of the reactor after a scram, (b) achieve subsequent cold shutdown of the reactor, and (c) maintain safe conditions during shutdown.

The standby liquid control (SLC) system is designed to provide an alternate method of reactor shutdown from full power to cold subcritical by the injection of a neutron absorbing solution to the RPV.

The feedwater control (FWC) system controls the flow of feedwater into the RPV to maintain the water level in the vessel within predetermined limits during all plant operating modes.

The neutron monitoring system (NMS) is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system is designed to provide indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. There are four subsystems in the NMS: the startup range neutron monitoring (SRNM) subsystem, the power range neutron monitoring (PRNM) subsystem [comprised of the local power range monitors (LPRM) and average power range monitors (APRM)], the automatic traversing in-core probe (ATIP) subsystem, and the multi-channel rod block monitoring (MRBM) subsystem.

Startup range neutron monitoring (SRNM) subsystem

The SRNM subsystem monitors the neutron flux from the source range to 15% of the rated power. The SRNM is designed to provide neutron flux related trip inputs (flux level and period) to the RPS, including a noncoincident trip function for refuelling operations and a coincident trip function for other modes of operation. The SRNM has 10 channels where each channel includes one detector installed at a fixed position within the core.

Power range neutron monitoring (PRNM) subsystem

The PRNM subsystem provides flux information for monitoring of the average power level of the reactor core. It also provides information for monitoring of the local power level. The PRNM is used when the reactor power is above approximately 1% of rated power.

The PRNM subsystem consists of two subsystems:

- Local power range monitoring (LPRM) subsystem,
- Average power range monitoring (APRM) subsystem.

The LPRM subsystem continuously monitors local core neutron flux. It consists of 52 detector assemblies with 4 detectors per assembly. The 208 LPRM detectors are separated and divided into four groups to provide four independent APRM signals. The APRM subsystem averages the readings of the assigned LPRM detectors and provides measurement of reactor core power. Individual LPRM signals are also transmitted through dedicated interface units to various systems such as the Reactor Control and Instrumentation System (RC&IS), and the plant process computer.

Automatic traversing in-core probe (ATIP) subsystem

The ATIP subsystem performs an axial scan of the neutron flux in the core at the LPRM assembly locations. The subsystem can be controlled manually by the operator, or it can be under micro-processor-based automated control. The ATIP subsystem consists of neutron-sensitive ion chambers, flexible drive cables, guide tubes, indexing machines, drive machines, and an automatic control system. Working in conjunction with the Performance monitoring and control system (PMCS), the ATIP subsystem calibrates the LPRM outputs.

Multi-channel rod block monitor (MRBM) subsystem

The MRBM subsystem is designed to stop the withdrawal of control rods and prevent fuel damage when the rods are incorrectly being continuously withdrawn, whether due to malfunction or operator error. The MRBM averages the LPRM signals surrounding each control rod being withdrawn. It compares the averaged LPRM signal to a preset rod block setpoint, and, if the averaged values exceeds this setpoint, the MRBM subsystem issues a control rod block demand to the RC&IS. The rod block setpoint is a core flow biased variable setpoint.

4.1.5 Electrical systems

4.1.5.1 Operational power supply systems

On-site power is supplied from either the plant turbine generator, utility power grid, or an off-site power source depending on the plant operating status. During normal operation, plant loads are supplied from the main generator through the unit auxiliary transformers. A generator breaker allows the unit auxiliary transformers to stay connected to the grid to supply loads by backfeeding from the switchyard when the turbine is not online.

Direct current power supply

The DC power supply system (DC) consists of three separate subsystems:

- safety related 125 V DC,
- non-safety related 250 V DC, and
- non-safety related 125 V DC.

The system begins at the source terminals of the plant safety and non-safety battery chargers. It ends at the input terminals of the plant DC loads (motor, control loads, etc.) and at the input terminals of the inverters of the low voltage vital AC power supply system.

Each DC subsystem consists of a battery, associated battery charger, power distribution panels, and all the associated control, monitoring and protective equipment and interconnecting cabling. In addition, DC employs standby chargers that are shared between the batteries to enable the individual battery testing and off-line equalization.

DC operates with its battery and battery chargers (except standby chargers) continuously connected to the DC system. During normal operation, the DC loads are powered from the battery chargers with the batteries receiving a continuous charging current (i.e., floating) on the system. In case of loss of AC power to the charger or its failure, the DC loads are automatically powered from the batteries.

Instrument and control power supply

The instrument and control power supply system (ICP) provides 120 V AC power to instrument and control loads which do not require continuity of power during a loss of preferred power.

The ICP system consists of class 1E and non-Class 1E interruptable power supplies and their respective regulating step-down power transformers (conditioners), a transfer switch (for non-class 1E subsystem only), alternating current (AC) distribution panels, and cables to the distribution system loads.

The ICP system is powered from 480V motor control centers (MCC) and is distributed at 208Y/120V. Power conditioners are used as voltage regulating transformers to regulate its output voltage to various I&C loads under broad variations in supply voltage and load changes. Power conditioners are sized to supply their respective I&C loads under the most demanding operating conditions.

4.1.5.2 Safety-related systems

Class IE AC power supply

The class 1E buses of the on-site power system consists of three independent divisions of class 1E equipment. Each division is fed by an independent class 1E bus at the medium voltage level, and each division has access to one on-site and two off-site (normal and alternate preferred) power sources. Each division has access to an additional power source which is provided by the combustion turbine generator (CTG).

Each division is provided with an on-site safety related standby diesel generator which supplies a separate on-site source of power for its division when normal or alternate preferred power is not available. The standby diesel generators are capable of providing the required power to safely shut down the reactor after loss of preferred power and/or loss of coolant accident and to maintain the safe shutdown condition and operate the class 1E auxiliaries necessary for plant safety after shutdown.

The on-site standby AC power supplies (diesel generators) have sufficient capacity to provide power to all their respective loads. Loss of the preferred power supply, as detected by undervoltage relays in each division, will cause the standby power supplies to start and automatically connect, in sufficient time to safely shut down the reactor or limit the consequences of a design basis accident (DBA) to acceptable limits and maintain the reactor in a safe condition.

Direct current power supply

The class 1E 125 V DC subsystem consists of four independent and redundant divisions (I, II, III, and IV). All four divisional batteries are sized to supply 125 V DC power to their loads during a design basis accident, coincident with loss of AC power, for a period of at least two hours based on the most limiting load profile without load shedding. This sizing of the division I battery also meets the requirement to permit operation of the station blackout coping systems for eight hours with manual load shedding. This manual load shedding commences only after the first two hours of station blackout and includes the vital AC power, as well as the remote multiplexing units (RMU) and division I Diesel generator control loads. The division I battery is sized to support operation of the Reactor Core Isolation Cooling (RCIC) System and remote shutdown system (RSD), as well as a minimum necessary emergency lighting. This manual load shedding takes credit for the RCIC operation from outside the main control room.

Vital (uninterruptable) power supply

The class 1E vital AC (VAC) power supply provides redundant, reliable power to the safety logic and control functions during normal, upset and accident conditions. The VAC is comprised of three independent subsystems. Each subsystem supplies uninterruptable, regulated AC power to those loads which require continuity of power during a loss of preferred power (LOPP).

Each VAC subsystem’s division or load group is comprised of an independent uninterupptable power supply, maintenance bypass switch, regulating transformers, main distribution panel, local distribution panels, and cables for power, instrumentation and control. Each uninterruptable power supply is a constant voltage constant frequency (CVCF) inverter power supply unit consisting of a rectifier, inverter, and AC and DC static transfer switches. Each CVCF power supply is provided with an alternate AC source with sufficient capacity to allow normal operation in case of failure or unavailability of a single inverter.

4.1.6 Safety concept

4.1.6.1 Safety requirements and design philosophy

Recognizing the need for continued safety enhancements in plant operation, one goal in designing the ABWR was to reduce core damage frequency by at least an order of magnitude relative to currently operating BWR plants. Essential design features contributing to this are enhancement of the high-pressure ECCS and RHR functions, including the emergency AC power supply, and the installation of diversified ATWS countermeasures. Furthermore, the adoption of reactor internal pumps (RIPs) eliminates large attached recirculation piping, particularly involving penetrations below the top of the core elevation, and make it possible for a smaller Emergency core cooling system (ECCS) network to maintain core coverage during a postulated loss of coolant accident.

4.1.6.2 Safety systems and features (active, passive and inherent)

The ABWR ECCS network was changed to a full three-division system, with both a high and low pressure injection pump and heat removal capability in each division. For diversity, one of the systems, the RCIC system (a safety-grade system in the ABWR), includes a steam driven high pressure pump. Transient response was improved by having three high pressure injection systems available in addition to feedwater. The adoption of three on-site emergency diesel-generators to support core cooling and heat removal, as well as the addition of an on-site gas turbine-generator reduces the likelihood of "station blackout." The balanced ECCS system has less reliance on the automatic depressurization system (ADS) function, since a single motor-driven High pressure core flooder (HPFL) is designed to maintain core coverage for any postulated line break size.

Response to anticipated transients without scram (ATWS) was improved by the adoption of the fine motion control rod drives (FMCRD), which allow reactor shutdown either by hydraulic or electric insertion. In addition, the need for rapid operator action to mitigate an ATWS was avoided by automation of emergency procedures, such as feedwater runback and standby liquid control (SLCS) injection.

4.1.6.3 Severe accidents (beyond design basis accidents)

The US ABWR also improved the capability to mitigate severe accidents even though such events are extremely unlikely. Through inerting, containment integrity threats from hydrogen generation were eliminated. Sufficient spreading area in the lower drywell, together with a passive drywell flooding system, assures coolability of postulated core debris. Manual connections make it possible to use on-site or off-site fire water systems to maintain core cooling. Finally, to reduce off-site consequences, a passive hard-piped wetwell vent, controlled by rupture disks set at twice design pressure (service level C), is designed to prevent catastrophic containment failure and provide maximum fission product “scrubbing.”

The result of this design effort is that in the event of a severe accident less than 0.25 Sv (25 rem) of radiation is released at the site boundary, even at a very low probability level. This means that the public's safety and health is assured. Figure 4.1-6 illustrates some of the severe accident mitigation features of the ABWR.

4.1.7 Plant layout

4.1.7.1 Buildings and structures, including plot plan

The ABWR plant includes all buildings which are dedicated exclusively or primarily to housing systems and the equipment related to the nuclear system or controls access to this equipment and systems. There are five such buildings within the scope:

(a) Reactor building - includes the reactor pressure vessel, containment, and major portions of the nuclear steam supply system, refuelling area, diesel generators, essential power, non-essential power, emergency core cooling systems, Heating, Ventilation and Cooling (HVAC) System and supporting systems.

(b) Service building - personnel facilities, security offices, and health physics station.
(c) Control building - includes the control room, the computer facility, reactor building component cooling water system and the control room HVAC system.
(d) Turbine building - houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.
(e) Radwaste building - houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

The site plan of the ABWR includes the reactor, service, control, turbine, radwaste and supporting buildings. Provision is made within the reactor building for 10 years spent fuel storage. Separate buildings can be provided for additional on-site waste storage and spent fuel storage for 20 years. Figure 4.1-7 illustrates the site plan of the ABWR.

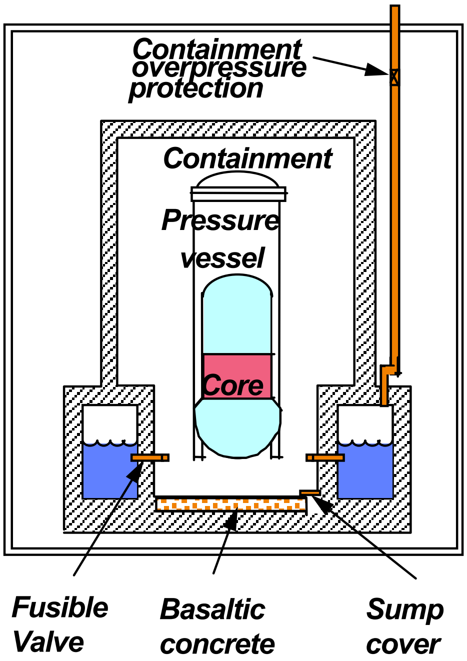

FIG. 4.1-6. ABWR - severe accident mitigation features.

Development of the ABWR plant and building arrangements has been guided by the following criteria:

(a) Retain the passive and well established BWR pressure suppression containment technology. Use of the horizontal vent configuration confirmed for the Mark III containments.
(b) Emphasize optimal layout of systems to improve personnel access and equipment maintenance activities.
(c) Locate major equipment for early installation using open top construction approach and large scale modularization.
(d) Arrange the reactor building around the primary containment to provide multiple barriers to post-accident fission product leakage, and high tolerance to external missiles.

The ABWR design arrangement minimizes material quantities. This, when combined with the volume reduction, contributes to the substantial reduction in both the construction schedule and plant capital cost.

The layout of the reactor and turbine buildings was based on the following considerations:

(a) Personnel access for all normal operating and maintenance activities was a primary concern starting with the first layout studies. Access routes from the change room to contaminated reactor and turbine building areas are as direct as possible. At each floor, 360° access is provided, if practical, to enhance daily inspections and normal work activities. Access to equipment not reachable from floor level is via platform and stair access wherever possible.
(b) Equipment access is provided for all surveillance, maintenance and replacement activities with local service areas and laydown space for periodic inspections. Adequate hallways and other equipment removal paths, including vertical access hatches, are provided for moving equipment from its installed position to service areas or out of the building for repair. Lifting points, monorails and other installed devices are provided to facilitate equipment handling and minimize the need for re-rigging individual equipment movements. The equipment access also considers the need for construction access.
(c) Radiation levels for personnel are controlled and minimized. The reactor building is divided into clean and controlled areas. Once personnel enter a clean or controlled area, it is not possible to crossover to the other area without returning to the change area. Redundant equipment is located in shielded cells to permit servicing one piece of equipment while the plant continues to operate. Valve galleries are provided to minimize personnel exposure during system operation or preparation for maintenance.

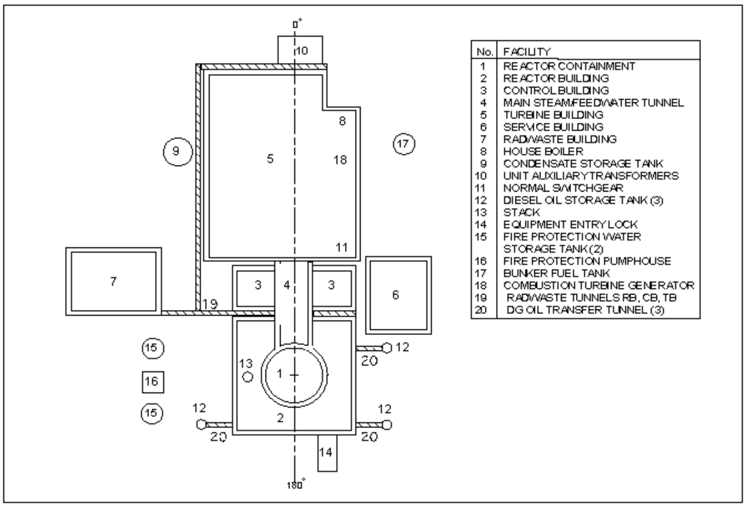

FIG. 4.1-7. ABWR - site plan

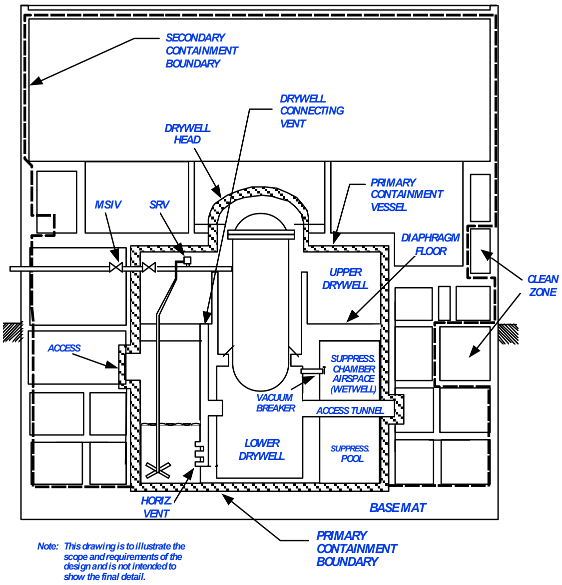

FIG. 4.1-8. ABWR - Containment structure features.

The turbine generator is aligned with its axis in-line with the reactor building. This is done to minimize the possibility of turbine missile impact on the containment vessel.

The main and auxiliary transformers are located adjacent to the main generator at the end of the turbine building. This location minimizes the length of the isophase bus duct between the generator and transformers, as well as the power supply cables back to the main electrical area of the power block.

The site plan includes consideration for construction access. The arrangement provides a clear access space around the reactor and turbine buildings for heavy lift mobile construction cranes without interference with other cranes, access ways and miscellaneous equipment.

4.1.7.2 Reactor building

The ABWR reactor building is a reinforced concrete structure. The integrated reactor building and containment structure has been analysed for a safe shutdown earthquake of 0.3g.

A secondary containment surrounds the primary containment and provides a second containment function including a standby gas treatment system. Off-site radiological dose studies have shown that a containment leak rate of less than 0.5%/day is achievable.

Careful attention has been given to ease of construction with this building arrangement. The building features full 360° access on all floors for ease of worker movement. Generally, the major cooling equipment has been placed on the lowest floors of the building to allow early installation during construction.

Modularization techniques are being implemented to reduce costs and improve construction schedules. These techniques will be applied to such reactor building items as (1) building reinforcing bar assemblies, (2) structural steel assemblies, (3) steel liners for the containment and associated water pools, and (4) selected equipment assemblies.

Removal of the post LOCA decay heat is achieved by the containment heat removal system, consisting of the suppression pool cooling mode, wetwell, and drywell and drywell spray features. An integral part of the RHR system, the system removes steam directly from the drywell and wetwell into the suppression pool. The large volume of water in the suppression pool serves as a fission product scrubbing and retention mechanism. The reactor building serves as an additional barrier between the primary containment and the environment. Any fission product leakage from the primary containment is expected to be contained within the reactor building.

Analyses of the radiological dose consequences for accidents, based on an assumed containment leak rate of 0.5% per day, show that the off-site doses after an accident is less than 1 rem. This favourable dose rate is made possible by trapping fission products within the secondary containment with a slight negative pressure and processing the air through the standby gas treatment system.

Key distinguishing features of the ABWR reactor building design include:

(a) Elimination of external recirculation loops reduces the containment volume associated with high construction costs.
(b) Reduced building volume reduces material costs and construction schedule.
(c) Designed with simple structural shapes to improve constructability to reduce capital costs and the construction schedule.
(d) Improved personnel and equipment access for enhanced operability and maintainability.

The volume of the ABWR reactor building has been reduced to approximately 167 000 cubic meters. Since this reduced volume was obtained by simplification of the reactor supporting systems and optimization of their arrangement with improved access (rather than simply by compaction), it provides material cost savings and helps reduce the construction schedule without adversely impacting maintenance.

The major equipment access to the reactor building is via double door vestibule at grade level. This entry area is connected to the refuelling floor by a large hatch serviced by the reactor building crane. The reactor building layout utilizes the grade level entry area for major servicing of the cooling equipment. All of the major pieces of equipment can be moved into the area through hatches.

4.1.7.3 Containment

The ABWR pressure suppression primary containment system comprises the drywell (DW), wetwell, and supporting systems. The main features of the ABWR containment structure are illustrated in Figure 4.1-8. A reinforced concrete containment vessel (RCCV) with an internal steel liner was adopted as the primary containment vessel (PCV) of ABWR. It is united with the reactor building (R/B) structure except for a drywell head and other penetrations. The steel RPV pedestal and the reinforced concrete diaphragm floor, partition the containment volume into a drywell and suppression chamber. The drywell and suppression chamber are connected by the steel vent pipes installed between a double shell steel structure of the RPV pedestal.

4.1.7.4 Turbine building

The turbine building houses all the components of the power conversion system. This includes the turbine-generator, main condenser, air ejector, steam packing exhauster, off-gas condenser, main steam system, turbine bypass system, condensate demineralizers, and the condensate and feedwater pumping and heating equipment. The small size of the ABWR turbine building makes a significant contribution to capital cost savings and a shorter construction schedule.

4.1.7.5 Other buildings

No information provided.

4.1.8 Technical data (for the US Version of ABWR *)

General plant data

Power plant output, gross	1385	MWe
Power plant output, net	1300	MWe
Reactor thermal output	3926	MWth
Power plant efficiency, net	33.1	%
Cooling water temperature	≈ 28.0	°C

Nuclear steam supply system

Number of coolant loops	1	
Primary circuit volume		m^3
Steam flow rate at nominal conditions	2122	kg/s
Feedwater flow rate at nominal conditions	2118	kg/s

Reactor coolant system

Primary coolant flow rate	14502	kg/s
Reactor operating pressure	7.07	MPa
Steam temperature/pressure	287.8/7.07	°C/MPa
Feedwater temperature	215.6	°C
Core coolant inlet temperature	278	°C
Core coolant outlet temperature	288	°C
Mean temperature rise across core	10	°C

Reactor core

Active core height	3.810	m (typical)
Equivalent core diameter	5.163	m
Heat transfer surface in the core	9142	m^2 (typical)
Average linear heat rate	13.3	kW/m (typical)
Fuel weight	157(appr.)	t U
Average fuel power density	25.0	kW/kg U
Average core power density	49.2	kW/l
Thermal heat flux, F_q	412	kW/m^2
Enthalpy rise, F_H	273	
Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	4470	mm
Rod array	10x10, square lattice	
Number of fuel assemblies	872	
Number of fuel rods/assembly	92 (full length78, partial length14)	
Number of spacers	8	
Enrichment (range) of first core, average	2.0 (appr.)	Wt%
Enrichment of reload fuel at equilibrium core	3 to 4	Wt%
Operating cycle length (fuel cycle length)	24	months
Average discharge burnup of fuel [capability]	>50,000	MWd/*t*
Cladding tube material	annealed, recrystallised Zr 2	
Cladding tube wall thickness	0.66	mm
Outer diameter of fuel rods	10.3	mm
Fuel channel/box; material	Zr-2	
Overall weight of assembly, including box	300	kg
Uranium weight/assembly	180(appr.)	kg
Active length of fuel rods	3.810	mm (typical)
Burnable absorber, strategy/material	axial and radial grading/ Gd_2O_3 mixed with fuel	
Number of control rods	205	
Absorber material	B_4C and Hafnium	
Drive mechanism	electro-mechanical/hydraulic	
Positioning rate 30	mm/*s*	
Soluble neutron absorber	Boron	

Reactor pressure vessel

Inner diameter of cylindrical shell	7 100	mm
Wall thickness of cylindrical shell	190	mm
Total height, inside	21 000	mm
Base material: cylindrical shell	low-alloy carbon steel	
RPV head	[to ASTM A533, grade B, ASTM A508, class 3, or equiv.]	
lining	stainless steel	
Design pressure/temperature	8.62/301.7	MPa/°C
Transport weight (lower part w/rigging)	1164	t
RPV head	≈100	t

Reactor recirculation pump

Type	variable speed, wet motor, single stage, vertical internal pump	
Number	10	
Design pressure/temperature	same as for RPV	MPa/^{o}C
Design mass flow rate (at operating conditions)	1453 (each)	kg/s
Pump head	0.287	MPa
Rated power of pump motor (nominal flow rate)	≈800	kW
Pump casing material	same as for RPV	
Pump speed (at rated conditions)	≤1500	rpm
Pump inertia		kg m^2

Primary containment

Type	Pressure-suppression/ reinforced concrete	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	29/36.1	m
Design pressure/temperature	310.3/171.1	kPa/^{o}C
Design leakage rate	0.5	vol%/day
Is secondary containment provided?	Yes	

Reactor auxiliary systems

Reactor water cleanup,	capacity	42.36	kg/s
	filter type	deep bed	
Residual heat removal,	at high pressure		kg/s
	at low pressure (100 ^{o}C)	253.8	MW
Coolant injection,	at high pressure (HPCF)	36.3	kg/s
	at low pressure (LPCF)	253.8	kg/s

Power supply systems

Main transformer,	rated voltage	27/(site condition)	kV
	rated capacity	1500	MVA
Plant transformers,	rated voltage	27/6.9/6.9	kV
	rated capacity	62.5	MVA
Start-up transformer	rated voltage	(site condition)/6.9	kV
	rated capacity	62.5	MVA
Medium voltage busbars (6 kV or 10 kV)		6.9	kV
Number of low voltage busbar systems		17 (typical)	
Standby diesel generating units: number		3	
	rated power	5	MW
Number of diesel-backed busbar systems		3	
Voltage level of these		6900	V AC
Number of DC distributions 6/1			
Voltage level of these		125/250	V DC
Number of battery-backed busbar systems		7	
Voltage level of these		120	V AC

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	six flow, tandem compound, single reheat	
Number of turbine sections per unit (e.g. HP/LP/LP)	HP/LP/LP/LP	
Turbine speed	1800	rpm
Overall length of turbine unit48 (appr.)		m
Overall width of turbine unit 9 (appr.)		m
HP inlet pressure/temperature	6.792/283.7	MPa/^{o}C

Generator

Type	3-phase, turbo-generator	
Rated power	1500	MVA
Active power	1385	MW
Voltage	27	kV
Frequency	60	Hz
Total generator mass, including exciter	730 (Typical appr.)	t
Overall length of generator	18.5 (Typical appr.)	m

Condenser

Type	shell type (3 shells)	
Number of tubes	1 tube pass/shell	
Heat transfer area	124,170	m^2
Cooling water flow rate	34.68	m^3/s
Cooling water temperature	depends on site condition	
Condenser pressure (HP shell)	11.75	kPa

Condensate pumps

Number	4 x 50%	
Flow rate	≈ 435	kg/s

Pump head	3.82	MPa
Temperature		^{o}C
Pump speed		rpm

<u>Condensate clean-up system</u>

Full flow/part flow		Full flow
Filter type	(deep bed or rod type)	deep bed

<u>Feedwater tank</u>

Volume	None	m^3
Pressure/temperature	-	MPa/ ^{o}C

<u>Feedwater pumps</u>

Number	$3 \times 65\%$	
Flow rate	≈ 1000	kg/s
Pump head	6	MPa
Feed pump power	10 (typical appr.)	MW
Feedwater temperature (final)	216	^{o}C
Pump speed	5000 (typical appr.)	rpm

<u>Condensate and feedwater heaters</u>

Number of heating stages,	low pressure	3×4
	high pressure	2×2
	feedwater tank	None

* Note) 4.1.8 shows the principal specifications of US ABWR design, as mentioned in 4.1.10.4. The core and fuel design show above is based on US ABWR design (10 x 10 fuel in US, 9 x 9 fuel in Japan on the other hand). The other design is almost similar to the world's first unit in Japan (Kashiwazaki-Kariwa Unit 6/7), except some site depend parameters which are not related to nuclear essential design.

4.1.9 Measures to enhance economy and maintainability

As aforementioned, the ABWR was designed to improve safety, operation and maintenance (O&M) practices, economics, radiation exposure and so on. This section points out which corresponding technology results in what kind of benefit category.

4.1.9.1 Design simplification

the Direct Cycle system of BWR is originally more simplified than dual cycle system of PWR. Furthermore, ABWR substitutes RIP (refer to 4.1.2.4) for large Primary Loop Recirculation pumps and piping of conventional BWR. This is an evolutionary simplified design which can condense the nuclear boiler system only within RPV attached no recirculation piping.

4.1.9.2 Operation flexibility improvement

Typical examples are FMCRD and the new-designed main control room to enhance the ABWR operation flexibility. The FMCRD is described in 4.1.2.2, and new-designed advanced main control room is in 4.1.4.1.

4.1.9.3 Cost reduction of equipment and structures

Equipment reduction (typical example):
As mentioned in 4.1.6.2, a full three-division system, with both a high and low pressure injection pump and heat removal capability in each division, is adopted, however capacity of ECCS is rather much reduced.

Structure reduction:
The Reactor Building volume is reduced to 167000 cubic meters, led by optimised equipment and piping arrangement and adoption of RCCV, then it provides material cost savings. (refer 4.1.7.2).

4.1.9.4 Reduction of construction period

Modularization techniques are the most effective for the short construction period of the ABWR (Refer 4.1.7.2) in addition to the building volume reduction aforementioned.

4.1.9.5 Scope reduction of the maintenance during operation and outages

The typical example of maintenance reduction is:
There are 103 HCUs for 205 FMCRDs, each HCU provides sufficient volume of water stored at high pressure in a pre-charged accumulator to scram <u>two</u> FMCRDs at any reactor pressure.

4.1.9.6 Making the maintenance easier and with lower radiation exposure

The typical example of easy maintenance:
The Sealless FMCRD eliminates the sealing parts where inspection and maintenance are most necessary, and also the seal detection system that requires monitoring during operation, thus making CRDs maintenance free, contributing to easier maintenance with lower radiation exposure. (Refer 4.1.2.2)

Lower radiation exposure:
Reduction in radioactive nuclide concentration in reactor water, control of radioactive nuclide deposition and optimization of the permanent radiation shielding result in radiation level reduction.

4.1.9.7 Increasing the capacity factor

Such total improvement results in actual ABWRs' excellent operating experiences. (Refer IAEA-TECDOC-1245, Ref. [2]).

4.1.9.8 Reduction of the power generating cost

The large, low-pressure turbine with a 52-inch last stage blade (LSB), the moisture separator/heater (MSH), and the heater drain pump up system, are adopted to increase the turbine system efficiency. (Refer 4.1.3.1) Furthermore, the drums of radioactive waste discharged are reduced by adoption of latest radioactive waste management system. (Refer IAEA-TECDOC-1175, Ref. [3]).

4.1.10 Project status and planned schedule

This section provides latest project status after the former report "IAEA-TECDOC-968"[1]. The status before 1997 is described in "IAEA-TECDOC-968".

4.1.10.1 Entities involved (Japan)

The Japanese version of the ABWR was developed by GE, Hitachi Ltd. and Toshiba Corp. under the sponsorship of the Tokyo Electric Power Company (TEPCO). In 1987, TEPCO announced its decision to proceed with a two-unit ABWR project at its Kashiwazaki-Kariwa nuclear power station, 220 kilometers northwest of Tokyo, as Kashiwazaki-Kariwa nuclear Power Station Unit 6&7. KK-6 and –7 began commercial operation in November 1996 and July 1997, respectively. Each is rated 1,315MWe (net). The results of the first ten reactor years of combined operating experience for TEPCO KK-6&7 indicated below:

- The ABWRs are performing up to expectations
- Unplanned shutdowns have been due to conventional problems and do not suggest there are any ABWR-specific problems
- BWR technology is becoming safer and more economic
- Compared to earlier BWR technology, ABWRs have lower occupational radiation exposure, increased availability, higher load factors and lower O&M costs
- ABWRs would operate more efficiently under less severe operating constraints and with improved management strategies

Ten ABWR deployment programmes are underway in Japan as follows:

* Hamaoka-5,	(commercial operation) c/o 2004 under construction (Chubu Electric Power Co. Inc.)
* Shika-2,	c/o 2005 under construction (Hokuriku Electric Power Co.)
* Shimane-3	c/o 2009 Site Authorized (Chugoku Electric Power Co.)
* Fukushima-I, 7&8	c/o 2008 & 2009 EIS submitted (TEPCO)
* Ohma (Full-MOX)	c/o 2009 Site Authorized (EPDC, Electric Power Development Corp.)
* Higashidori-1&2	c/o 2010 & later Planned (TEPCO)
* Kaminoseki 1&2	c/o 2012 & 15 Site Authorized (Chugoku Electric Power Co.)

FIG. 4.1-8. Kashiwazaki-Kariwa Nuclear Power Station Unit 6&7 (Tokyo Electric Power Co.) (Right Photo: from left, Unit 7, Unit 6 (ABWR, 1356MWe) and Unit 5 (BWR-5, 1100MWe)).

4.1.10.2 Entities involved (US)

First of a kind engineering (FOAKE) has been conducted for the ABWR design for application in the United States. Funding for the ABWR FOAKE project was provided by GE and its FOAKE associates including members of the ABWR FOAKE design team, the advanced reactor Corporation (ARC), representing utility sponsors of the ABWR FOAKE project, and the United States Department of Energy (DOE). The ABWR FOAKE Project began in June 1993 and was completed in 1996.

The result of FOAKE and Design Certificate from USNRC makes it possible to construct ABWRs in the US from the viewpoint of preliminary engineering and reactor type certified. In fact, it was also reflected to the Taiwan, China project as below.

4.1.10.3 Entities involved (Taiwan, China)

Through a competitive bidding process, Taiwan Power Co. (TPC) selected the ABWR for its two unit Lungmen project. GE will design and provide the scope of supply for two 1,350 MWe ABWRs. The Lungmen project will also be supported by the GE Team including: Black & Veatch, Hitachi, Shimizu, Toshiba, and other US, Taiwan, China, and international participants. Commercial operation for the two units is scheduled for 2006 and 2007.

4.1.10.4 Design status and Licensing process in US

On September 29, 1987, GE applied for certification of the U.S.ABWR standard design with the US Nuclear Regulatory Commission (USNRC). The USNRC staff issued a final safety evaluation report (FSER) related to the certification of the U.S.ABWR design in July 1994 (NUREG-1503). The FSER documents the results of the NRC staff's safety review of the U.S.ABWR design against the requirements of 10 CFR Part 52, Subpart B, and delineates the scope of the technical details considered in evaluating the proposed design. Subsequently, the applicant submitted changes to the U.S.ABWR design and the NRC staff evaluated these design changes in a supplement to the FSER (NUREG-1503, Supplement No.1).

USNRC adopted as final this design certification rule, Appendix A to 10 CFR Part 52, for the U.S. ABWR design in May 1997.

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4.2 ABWR-II (GENERAL ELECTRIC, USA / HITACHI LTD. AND TOSHIBA CORP., JAPAN)

ABWR-II, the evolutionary reactor based on the advanced BWR (the ABWR), is now under development, jointly by the six Japanese BWR utilities led by Tokyo Electric Power Company (TEPCO), General Electric Company, Hitachi Limited, and Toshiba Corporation. This project to develop a next generation reactor was launched in the early 1990s when the first ABWR was still under construction at Kashiwazaki-Kariwa. Initiating this project was not considered premature since replacement of operating power plants were anticipated in the next twenty years and sufficient lead-time was required to develop a new reactor.

The delay of FBR development has bolstered up the role of light water reactors, and deregulation of the electric power business highlighted the urgency of improving economics of nuclear power generation. For these reasons, economical competitiveness became one of the most important objectives of developing ABWR-II, while achieving the highest standards of safety was another important objective. In order to lessen R&D cost, the project focused on improving the current ABWR design rather than pursuing revolutionary technologies, but succeeded in coming up with a design compatible from both economical and safety points of view.

4.2.1 Introduction

By adopting a large electric output (1700 MWe), a large fuel bundle, a modified ECCS, and passive heat removal systems, and other design features, a design capable of increasing both economic competitiveness and safety performance is achieved. The key objectives of ABWR-II are further improvement in economics against alternative forms of generation and enhancement of safety & reliability. The design goals are:

- Economic competitiveness
  - Power generation cost: 20% cost reduction from standardized ABWR
  - Overnight capital cost: 30% cost reduction from standardized ABWR
  - Construction period: 29.5 months (from first concrete work to fuel loading)
- Safety and reliability
  - Good combination of active and passive systems
  - Provision of grace period both for transients and accidents: one day grace period
  - Consideration of severe accident from design stage
  - Refinement of PSA performance (equal to or higher than that of ABWR, especially on containment capability)
- Sustainability for future fuel cycle uncertainty
  - Increased flexibility for higher burn-up, MOX and higher conversion

4.2.2 Description of the nuclear systems

4.2.2.1 Primary circuit and its main characteristics

The primary functions of the ABWR-II nuclear boiler system are:

(1) To deliver steam from the reactor pressure vessel (RPV) to the turbine main steam system;
(2) To deliver feedwater from the condensate and feedwater system to the RPV;
(3) To provide overpressure protection of the reactor coolant pressure boundary;
(4) To provide automatic depressurization of the RPV in the event of a loss of coolant accident (LOCA) where the RPV does not depressurize rapidly; and
(5) With the exception of monitoring conditions in the RPV such as RPV pressure, metal temperature, and water level instrumentation.

Main steam lines (MSLs) are designed to direct steam from the RPV to the main steam system of the turbine, and feedwater lines to direct feedwater from condensate and feedwater system to the RPV.

The main steam line flow limiter, a flow restricting venturi built in the RPV MSL nozzle of each of the four main steam lines, limits the coolant blowdown rate from the reactor vessel to a choked flow rate equal to or less than 200% of rated steam flow at 7.07 MPa upstream gauge pressure in the event when a main steam-line break occurs anywhere downstream of the nozzle.

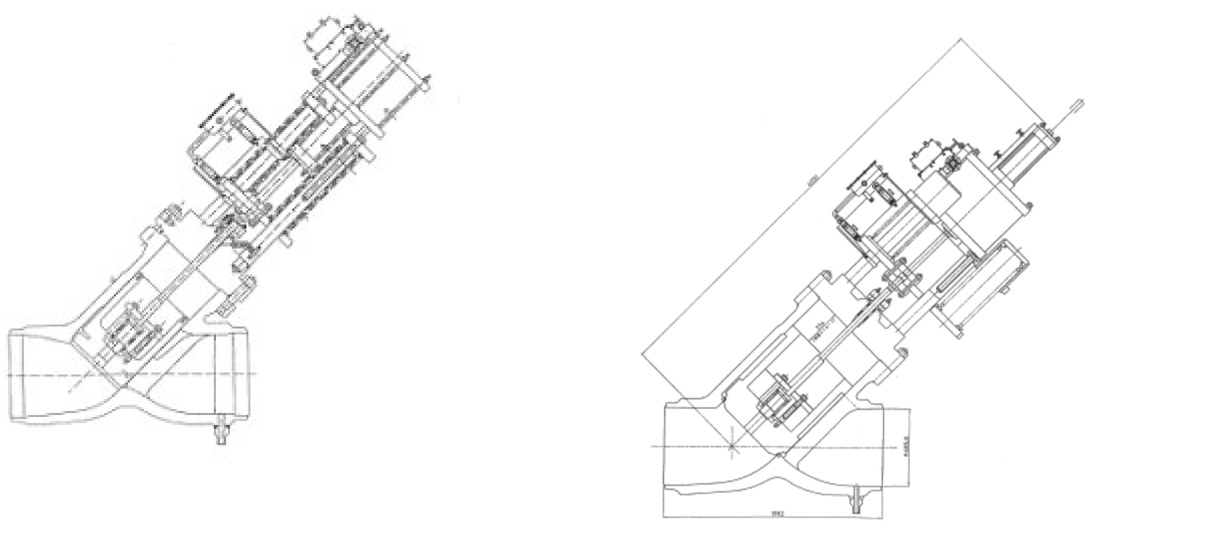
There are two main steam isolation valves (MSIVs) welded into each of the four MSLs: one MSIV inside the containment and one MSIV outside the containment. The MSIVs are Y-pattern globe valves. The Y-pattern configuration permits the inlet and outlet flow passages to be streamlined to minimize pressure drop during normal steam flow.

The new type MSIV for ABWR-II, now under development, is illustrated in Figure 4.2-1. The bore diameter is increased and the center of gravity of its driving mechanism is lowered. This bore diameter increase is not simply an enlargement from ABWR but is optimized so that the pressure loss is decreased to increase plant efficiency. The lowered center of gravity of the driving mechanism by relocating springs and an oil damper will contribute to improvement on seismic capability.

The nuclear pressure relief system consists of safety/relief valves (SRVs) located on the MSLs between the RPV and the inboard MSIV. There are 14 SRVs distributed on the four MSLs. The SRVs are designed to provide three main protection functions: overpressure safety, overpressure relief, and depressurization operation, which is discussed below separately.

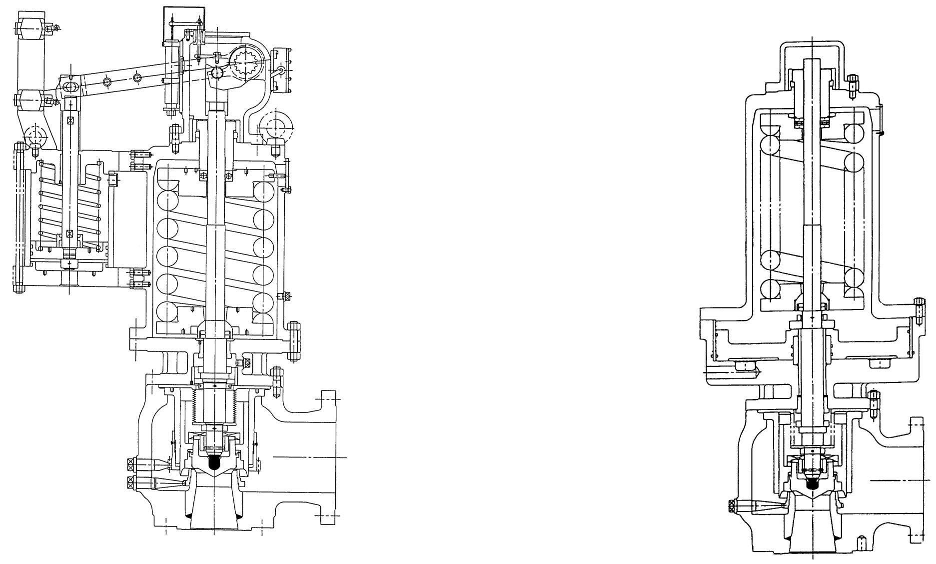
For ABWR-II, an increase of discharge capacity and simplification of valve structure were considered. To decrease the number of SRVs, the discharge capacity per SRV is increased by 70 % to 680 t/h (nominal) from ABWR's 395 t/h with an increased throat diameter and increased coil spring diameter. Also, the structure of the SRV is simplified by integrating an air cylinder into the SRV's main body, as illustrated in Figure 4.2-2. The development test programme for the new SRV is proceeding.

The automatic depressurization subsystem (ADS) consists of the six SRVs and their associated instrumentation and controls. The ADS designated valves open automatically for events involved with small breaks in the nuclear system process barrier or manually in the power actuated mode when required. The ADS designated valves are capable of operating from either ADS LOCA logic or overpressure relief logic signals. The ADS accumulator capacity is designed to open the SRV against the design drywell pressure following failure of the pneumatic supply to the accumulator.



ABWR MSIV *ABWR-II MSIV*

FIG. 4.2-1. MSIVs for ABWR and ABWR-II.



ABWR SRV ABWR-II SRV

FIG. 4.2-2. SRVs for ABWR and ABWR-II.

4.2.2.2 Reactor core and fuel design

Reference design

Basic policies of the ABWR-II core design are as follows:

- Keep the same level margin as the current core design in order to have enough flexibility for future higher burnup and longer cycle operation, under the condition to meet power uprate to 1700 MW(e);
- Reduce components and shorten the refueling outage time to improve the capacity factor.

To meet above requirements, 1.5 times larger fuel bundle and K-lattice were selected for the reference core design. The large bundle is able to increase the area inside the channel box and has potential for increasing the number of fuel rods in support of uprating plant output. On the other hand, the fuel bundle pitch increase results in less cold shutdown margin (CSDM) due to decrease of the number of CRs. The K-lattice control rod concept was chosen as a countermeasure for cold shutdown margin. This concept, compared to the conventional control rod design (N-lattice), is illustrated in Figure 4.2-3. In the K-lattice concept, the number of CRs per bundle is increased to two CRs for every four fuel bundles, while there is one CR per four bundles for the conventional lattice. So the K-lattice control concept provides improvement in CSDM and makes it possible to adopt a larger bundle.

The specifications of the ABWR-II core are listed in Table I and the configuration of the core is displayed in Figure 4.2-4. The thermal output of the ABWR-II core is 4960 MW, 1.26 times larger than that of the ABWR core. Although the former has 424 bundles, which is 49% of that of the ABWR, there are 197 control rods (CRs), about the same as in the ABWR. This is because the ABWR-II employs the K-lattice in order to maintain the cold shutdown margin (CSDM). However, the ABWR-II has fewer CRs per unit output than the ABWR.

Targeted operating cycle length and average discharge burnup are set to be 18 EFPM and 60GWd/t, respectively. The recirculation flow control range of 20% is achieved.

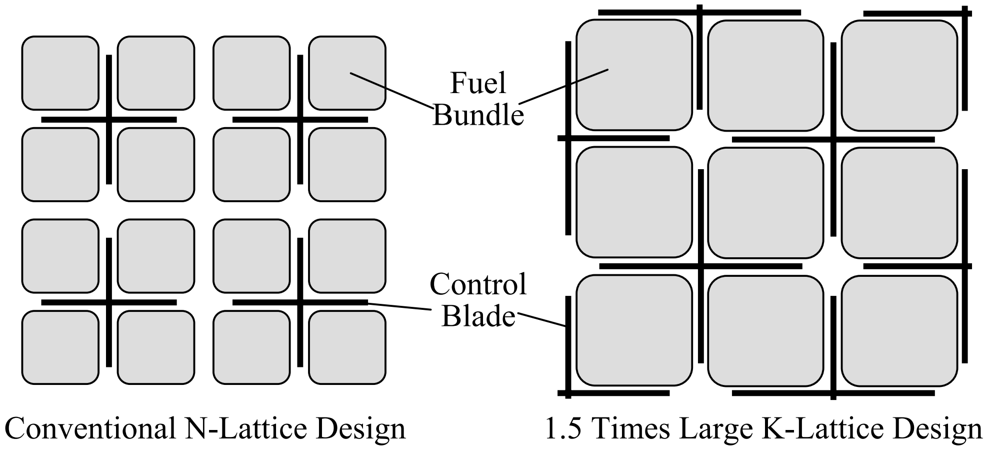


FIG. 4.2-3. Comparison of Lattice Configurations

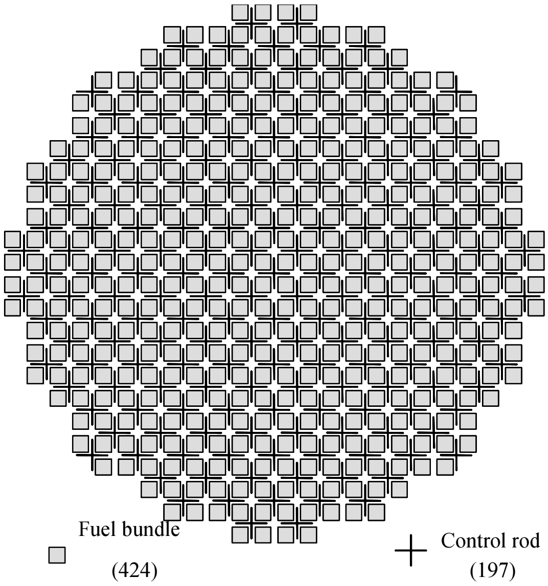
TABLE 4.2-I. MAIN PARAMETERS FOR ABWR-II CORE

-

Ite	Un	ABW	AB
Electric	M	170	135
Reactor	M	496	392
Operating	EFP	1	1
Average	GW	6	4
Maximum core	t/h	$62.1x^3$	$57.9x^3$
Active core	m	3.7	3.7
Fuel bundle	c	23.	15.
Number of	-	19	20
Number of fuel	-	42	87

The main characteristic of the ABWR-II bundle is 1.5 times larger bundle pitch compared to the conventional BWR bundle. This large bundle comprises four sub-bundles, each of which consists of an 8 x 8 fuel arrangement.

In the reference design, shown in Figure 4.2-5, the four sub-bundles are separated by a partition and there is a large water box in the center, formed by parts of the partition, which occupies 24 fuel rod positions. In addition, there are eight small water rods, each equivalent in size to the fuel rod.



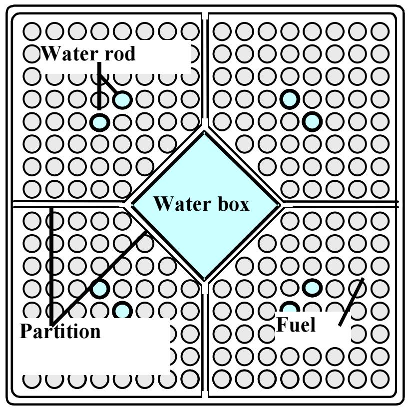


FIG. 4.2-4. ABWR-II core configuration

FIG. 4.2-5. Reference bundle configuration.

Spectral shift rod (SSR)

In the BWR, the excess reactivity is controlled by burnable poison, CRs and recirculation flow. The recirculation flow can control the reactivity through the void fraction, which changes the neutron spectrum in the core. In order to reduce use of CRs under ordinary operation and improve the neutron economy, it is desirable to enlarge the controllable reactivity using recirculation flow. The reference core design has a more negative void coefficient due to larger fuel inventory and reactivity control capability with core flow becomes larger than current core design.

In order to enlarge the reactivity change due to unit recirculation flow change further, the SSR has been developed as an option on core and fuel design. The SSR is a component to be used instead of the water rod, in which the water level develops naturally during operation and changes according to the recirculation flow rate through the channel. The SSR configuration and mechanism, illustrated in Figure 4.2-6, consists of a large ascending path and a narrow descending path. The coolant enters the ascending path from below the lower tie plate (LTP), goes up to the top, comes down in the descending path, and goes out right above the LTP. With this structure and a tight inlet, the water flow rate into the SSR is so small that the water heated by the irradiation of gamma rays and neutrons reaches the saturation temperature and boils in the SSR. Since the water velocity is very small in the ascending path, a water level develops there. In this system, the pressure drop caused by the coolant passing through the LTP almost equals the main part of the static head of the water column in the ascending path, which is the part from the height of the exit hole to the height of the water level. The pressure drop at the LTP is nearly proportional to the second power of the flow rate through it. Thus, by changing the recirculation flow rate, the water level in the SSR can be varied substantially.

Water shifting of the SSR also affects the average void fraction in the operation. Earlier in the cycle, the water level in the SSR is low, hence the peak of axial power shape tends to be located at the bottom half, where more water is present. Usually the average void fraction earlier in the cycle is higher due to the low recirculation flow rate. Then this axial power shape enhances that. Then, later in the cycle as the water comes up in the SSR, the peak of the axial power moves to the upper half, which makes the average void fraction much lower. By a synergistic effect of the water level change in the SSRs and an enhanced void fraction change in the channels, the reactivity change by the recirculation flow rate can be enlarged. In addition, fuel utilization is improved by the enhanced spectral shift operation, namely a harder neutron spectrum at BOC and a softer neutron spectrum at EOC than ordinary BWR operation.

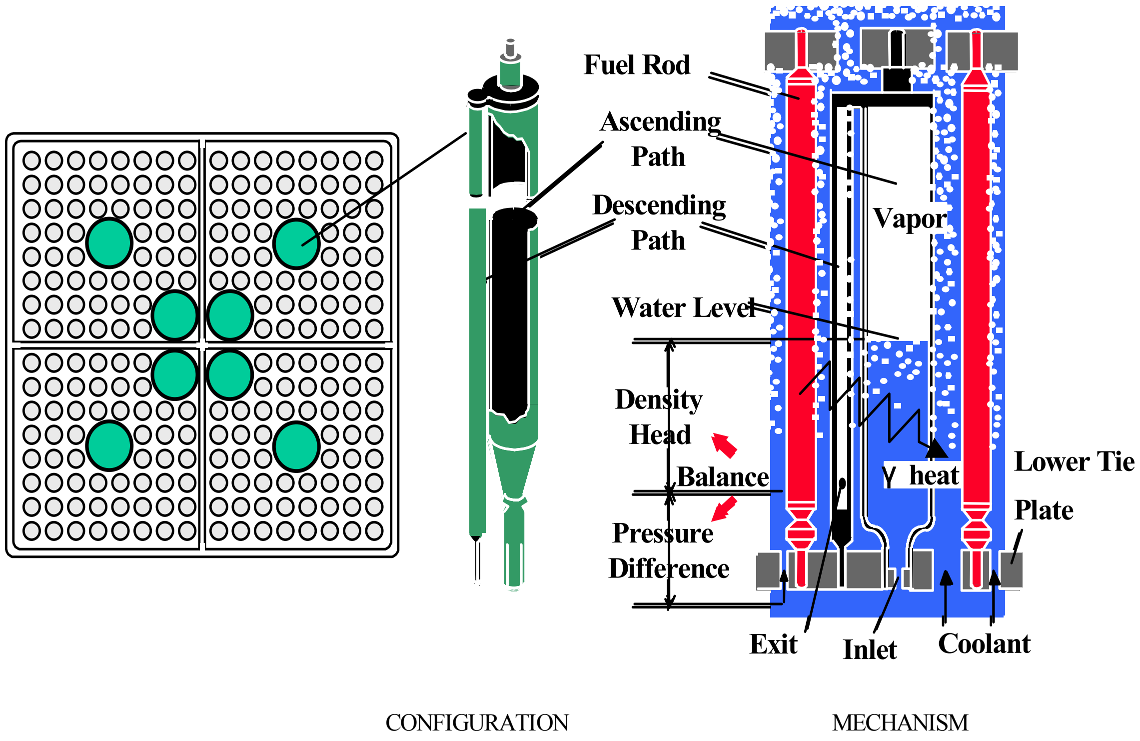


FIG 4.2.6. Configuration and operational principle of SSR

In the case of the ABWR-II assembly, which has a larger in-channel space, a larger volume fraction of the SSR is affordable compared with the conventional BWR assembly. The results of the equilibrium cycle analyses with 18-month cycle operation and discharge burnup of 60GWd/t showed that the SSR core design could allow operation with all CRs withdrawn throughout the entire cycle without increasing the maximum core flow rate. It was also shown that the uranium saving factor of about 6-7% against the reference ABWR-II core could be expected due to the higher spectral shift effect and lower Gd enrichment design. By incorporating the SSRs into the ABWR-II, the capacity factor can be increased because the necessity for CR and CR drive inspections is lessened. Moreover, the operation becomes much easier due to no need for a CR strategy and operation. The combination of these advantages with higher fuel utilization means the ABWR-II with SSRs should be an attractive alternative for the next generation nuclear reactor.

Control rod drive system

The control rod drive (CRD) system is composed of three major elements: the fine motion control rod drive (FMCRD) mechanisms; the hydraulic control unit (HCU) assemblies, and the control rod drive hydraulic (CRDH) subsystems.

Compared with ABWR's design, an improvement in the connection between the CRD motor and the CRD shaft was made. The improved mechanism is called a magnetic coupling. The magnetic coupling can transmit torque between the CRD motor and the CRD shaft through the pressure boundary without a penetration instead of having a seal around a penetrating shaft. This improvement eliminates the sealing parts where inspection and maintenance are most necessary, and also the seal detection system that requires monitoring during operation, thus making CRDs maintenance free, contributory to maintenance cost reduction.

In addition, the improved HCU, which is under development, will be adopted to ABWR-II. One improved HCU provides sufficient volumes of water to scram three FMCRDs at any reactor pressure. In case of conventional HCUs,.99 HCUs are required for 197 CRs, By applying the improved ones,65 improved HCUs and one conventional HCUs are enough for them.

4.2.2.3 Fuel handling and transfer systems

The reactor building is supplied with a refueling machine for fuel movement and servicing. The fuel handling and transfer system for ABWR-II is basically the same as on ABWRs except the fuel assembly weight of ABWR-II is heavier than that of ABWR.

4.2.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel (RPV) system consists of (1) the RPV and its appurtenances, supports and insulation, and (2) the reactor internal components enclosed by the vessel, excluding the core (fuel assemblies, control rods, in-core nuclear instrumentation and neutron sources), reactor internal pumps (RIPs), and control rod drives (CRDs). The RPV system is located in the primary containment. The reactor coolant pressure boundary portion of the RPV and its appurtenances act as a radioactive material barrier during plant operation.

Certain reactor internals support the core, flood the core during a loss of coolant accident (LOCA) and support safety related instrumentation. Other RPV internals direct coolant flow, separate steam, hold material surveillance specimens, and support instrumentation utilized for plant operation.

The RPV system provides guidance and support for the CRDs. It also distributes sodium pentaborate solution when injected from the standby liquid control (SLC) system.

The RPV system restrains the CRD to prevent ejection of the control rod connected with the CRD in the event of a failure of the reactor coolant pressure boundary associated with the CRD housing weld. A restraint system is also provided for each RIP in order to prevent the RIP from becoming a missile in the event of a failure of the reactor coolant pressure boundary associated with the RIP casing weld.

The RPV is a vertical, cylindrical vessel of welded construction with removable top head and head closure bolting seals. Trough the use of large forged rings, the number of welds in the RPV is reduced. The main body of the installed RPV has a cylindrical shell, flange, bottom head, RIP casings, penetrations, brackets, nozzles, and the shroud support, which has a pump deck forming the partition between the RIP suction and discharge. The shroud support is an assembly consisting of a short vertical cylindrical shell, a horizontal annular pump deck plate and vertical support legs.

An integral reactor vessel support skirt supports the vessel on the reactor pressure vessel pedestal. Anchor bolts extend from the pedestal through the flange of the skirt. RPV stabilizers are provided in the upper portion of the RPV to resist horizontal loads. Lateral supports for the CRD housings and in-core housings are provided.

Reactor internals

The major reactor internal components in the RPV system are: (1) core support structure, and (2) other reactor internals.

The core support structures encompass: the shroud, shroud support and a portion of CRD housings inside the RPV, core plate, top guide, fuel support, and control rod guide tubes (CRGTs).

Other reactor internals are:

- Feedwater spargers, shutdown cooling (SDC) and low pressure core flooder (LPFL) spargers for the residual heat removal (RHR) system, high pressure core flooder (HPCF) spargers and couplings, and a portion of the in-core housings inside the RPV and in-core guide tubes (ICGTs) with stabilizers;
- Surveillance specimen holders, shroud head and steam separators assembly and the dryer assembly.

Reactor recirculation pumps

The reactor recirculation system features an arrangement of ten variable speed reactor coolant recirculation pumps. The pumps with motors are mounted in the bottom of the RPV, and are thus termed reactor internal pumps (RIPs). The RIPs provide forced circulation of the reactor coolant through the lower plenum of the reactor and up through the lower grid, the reactor core, steam separators, and back down the downcomer annulus.

The recirculation flow rate is variable over a “flow control range”, from minimum flow established by certain pump performance characteristics to above maximum flow required to obtain rated reactor power.

Each RIP includes a device which prevents reverse RIP motor rotation by reverse flow induced torque. The RIP motor cooling is provided by an auxiliary impeller mounted on the bottom of the motor rotor, which circulates water through the RIP motor and its cooling heat exchanger. The heat exchangers are cooled by the reactor building closed cooling water (RCW) system.

4.2.2.5 Reactor auxiliary systems

The main auxiliary systems in the ABWR-II nuclear island consist of the residual heat removal (RHR) system, the reactor building closed cooling water (RCW) system, the reactor building seawater (RSW) system, the reactor water cleanup (CUW) system, the fuel pool cooling and cleanup (FPC) system and the suppression pool cleanup (SPCU) system. In addition there are many other auxiliary systems such as instrument and service air, condensate and demineralized water transfer, HVAC, equipment drain, floor drain and other systems which are basically the same as on ABWR plants and are not covered here.

Optimization was made in RHR system together with RCW and RSW systems. Taking into consideration that the passive heat removal systems of ABWR-II can be counted as a backup, the basic system configuration of RCW is two divisions instead of the three in ABWR. For RHR, RSW and active components in RCW in total make up four-division configuration that facilitates on-line maintenance and increases reliability and safety.

As to emergency power sources for active components in RHR/RCW/RSW systems, a four-division configuration consisting of two diesel generators and two gas turbine generators is applied to increase diversity and to facilitate maintenance. On-line maintenance will be applied to the diesel generators. The gas turbine generators are expected to be maintenance free.

The CUW system consists of piping, valves, pumps, heat exchangers and filter demineralizers which are used to remove impurities from the reactor primary coolant water to maintain water quality within acceptable limits during the various plant operating modes. The CUW design for ABWR-II is basically the same as on ABWRs.

The FPC and SPCU systems consist of piping, valves, pumps, heat exchangers and filter demineralizers which are used to remove decay heat from the spent fuel storage pool and to remove impurities from the water in the spent fuel pool and dryer/separator pool and suppression pool to maintain water quality within acceptable limits during the various plant operating modes. The FPC and SPCU systems are basically the same as on ABWRs.

4.2.2.6 *Operating characteristics*

No information provided.

4.2.3 Description of turbine generator plant systems

4.2.3.1 *Turbine generator plant*

The main turbine

The main turbine is six flow, tandem compound, reheat 1500 rpm machine with 1320.8mm (52 in.) last stage blades. The turbine has one duel-exhaust high pressure section and three dual-exhaust low pressure sections. The cycle uses moisture separators and reheaters with reheat for the cross-around steam.

Extraction steam from the high and low-pressure turbine extraction nozzles is conveyed to the high and low-pressure feedwater heaters, respectively. The feedwater heating systems are designed to provide a final feedwater temperature 216°C(420°F) at 100 percent nuclear boiling rate. This cycle yields a gross generator output of approx. 1700 MW with a thermal reactor output of 4960 MW.

Turbine bypass system

The turbine bypass system (TBP) provides the capability to discharge main steam from the reactor directly to the condenser to minimize step load reduction transient effects on the reactor coolant system. The TBP is also used to discharge main steam during reactor hot standby and cooldown operations. The TBP consists of valve chest that is connected to the main steam lines upstream of the turbine stop valves, and dump lines that separately connect each bypass valve outlet to condenser shell.

Main condenser

The main condenser, which does not serve or support any safety function and has no safety design basis, is a three-shell type deaerating type condenser. During plant operation, steam expanding through the low pressure turbines is directed downward into the main condenser and condensed. The main condenser also serves as a heat sink for the turbine bypass system, emergency and high level feedwater heater and drain tank dumps, and various other startup drains and relief valve discharges.

Each condenser shell has two tube bundles. Circulating water flows through the three shells. The main condenser is located in the turbine building in pits below the operating floor and is supported by the turbine building base mat.

4.2.3.2 *Condensate and feedwater systems*

The condensate and feedwater system are designed to provide a dependable supply of high-quality feedwater to the reactor at the required flow, pressure, and temperature. The condensate pumps take the deaerated condensate from the main condenser hotwell and deliver it through the steam jet air ejector condenser, the gland steam condenser, the condensate demineralizer, and through three parallel strings of low pressure feedwater heaters to the reactor feed pumps' section. The reactor feed pumps discharge through two stages of high pressure heater (two parallel strings) to the reactor.

4.2.3.3 *Auxiliary systems*

The turbine building cooling water system (TBCW), which is a non-safety related system, removes heat from the auxiliary equipment in the turbine building and rejects this heat to the turbine building service water system (TBSW) system.

4.2.4 Instrumentation and control systems

Basically, a high-reliability design comparable to that of ABWRs is adopted for the ABWR-II. That is, integrated digitalized system is applied in the I&C of ABWR-II. It should be noted, however, that some of the instrumentation and control systems have been more sophisticated and highly reliable from the view point of its realizibility and effectivity as described below:

(1) Advanced man-machine interface

On investigation of regular outage maintenance of the preceding ABWRs (K-6/7), some improvements to the ABWR plant man-machine interface have been pointed out as for various isolations and monitoring operation for inspection.

That is, the ABWR panels are good enough for start-up, shut-down and usual operation, however, as for regular outage some improvement needs as below have been mentioned, and shall be applied to the ABWR-II plant man-machine interface:

a. Efficiency of assistant operator and inspection team;
b. Visualization of inspection and maintenance status;
c. Judgment support of work conditions;
d. Efficiency of system inspection; and
e. Improvement of the I&Cmaintainability.

(2) Advanced control system

A symptom based transient mitigation method for feedwater system failure has been developed. This method will be able to monitor plant primary parameters and control other systems, if necessary, as the top plant control system in order to avoid unnecessary plant shut-down, controlling reactor water level.

This representative system is the automatic power output adjustable device. This is the system that outputs operating signals of the recirculation flow and control rods, etc. and controls the reactor power in case of feedwater system failure, based on various information on the core status and the control rods.

(3) Transient mitigation system

This system has been developed to reduce ΔMCPR for the most severe transient as below:

(a) Power load shut down bypass valve failure (ΔMinimum Critical Power Ratio (MCPR)=0.13)
(b) Loss of feedwater heating (ΔMCPR=0.14)

To reduce these ΔMCPR, this system is equipped with the function to open the relief valve as soon as detecting the power load shut down bypass valve failure and also to activate the selected control rods insert or RIP run back signals as soon as detecting loss of feedwater heating from a change of feedwater temperature.

As a result of applying this system, ΔMCPR can be reduced to 0.08 for item (a) and 0.10 for item (b).

(4) Gamma thermometer

Gamma thermometer is activated by the principle that gamma thermometer rod shall be heated with nuclear fission and its generated gamma rays. Generated heat is proportioned with surrounding fuel rods power density and flows to the coolant residual heat removal systems along to the heat conducting pass. This heat along the conducting pass shall be measured by the thermo-couple and its signal is a result of the proportion to the core power output.

This system is applied for a correcting system of the core power.

(5) Advanced sensor technology

At first, the following investigations on the present process instrumentation methods of the nuclear plant and study on the application of ray technology based sensing or transmission method have been done:

a. Investigation on the present sensor
b. Investigation on the field bass method
c. Investigation on the ray fiber dyne method
d. Investigation on ray sensing instrumentation

As a result, ray fiber instrumentation system has been selected for ABWR-II design.

4.2.4.1 Design concepts, including control room

Description of major design features of the ABWR-II included in this category is outlined as a part of the paragragh 4.2.4. The other design features except those mentioned in the paragraph 4.2.4 are basically the same as those of the ABWR.

4.2.4.2 Reactor protection and other safety systems

Description of major design features of the ABWR-II included in this category is outlined as a part of paragraph 4.2.4. The other design features except those mentioned in the paragraph 4.2.4 are basically the same as those of the ABWR.

4.2.5 Electrical systems

Basically the high-reliability design, which is comparable to that of the ABWR, is adopted for the auxiliary electrical power supply systems of the ABWR-II. However, there are three major differences between that of ABWR and that of ABWR-II:

(1) The auxiliary normal electrical power supply systems of ABWR-II have partly applied the 11.5kV high voltage normal buses which feed power to the large capacity IMs such as RFPs of the 1700MWe class plant. Those of ABWR have four 6.9kV high voltage buses.

(2) The on-site emergency power supplies of ABWR-II are composed of two gas turbine driven generators and two diesel driven generators. Those of ABWR are composed of three diesel driven generators

(3) The number of the safety-related buses of ABWR-II is four. That of ABWR is three.

4.2.5.1 Operational power supply systems

Description of major design features of the ABWR-II included in this category is outlined as a part of the paragraph 4.2.5. The other design features except those mentioned in the paragraph 4.2.5 are basically the same as those of the ABWR.

4.2.5.2 Safety-related systems

Description of major design features of the ABWR-II included in this category is outlined as a part of the paragraph 4.2.5. The DC power supplies of ABWR-II are also four divisions.

4.2.6 Safety concept

4.2.6.1 Safety requirements and design philosophy

The safety related requirements established during early phases of ABWR-II development are:

- Good combination of active and passive systems;

- Provision of grace period both for transients and accidents;
- Consideration of severe accident from design stage;
- Refinement of PSA performance (equal to or higher than that of ABWR, especially on containment capability).

Considering these requirements, ABWR-II design provides more emphasis on beyond-DBA capability in order to achieve a high level of safety such as the practical exclusion of the probability of emergency evacuation/resettlement. Optimization of safety and economic aspects is also to be strongly pursued. In order to accomplish these objectives, the following design approach was taken:

- Systems important to safety, are incorporated in an integrated manner;
- Hardware increase is minimized for cost dominant portion;
- Additional benefits are introduced, as much as possible.

The safety related system configurations and their performance are described in the following sections.

4.2.6.2 Safety systems and features (active, passive and inherent)

The current reference concept includes the following safety related system design features:

- Rationalized four division RHR
- Diversified emergency power supply
- Reactor Core Isolation Cooling (RCIC) System with a generator
- Passive heat removal systems

The ABWR-II ECCS configuration is shown in Figure 4.2-7. Cooling water injection system is comprising from high pressure core flooder (HPCF) and low pressure flooder (LPFL). Optimization was made in RHR system together with reactor building closed cooling water (RCW) system and reactor building seawater (RSW) system. Taking into consideration that the passive heat removal systems of ABWR-II can be counted as a backup, the basic system configuration of RCW is two division instead of the three in ABWR. This two-division configuration is expected to reduce equipment cost for RCW that has relatively large amount of materials especially for piping. For RHR, RSW and active components in RCW in total make up four-division configuration that facilitates on-line maintenance and increases reliability and safety.

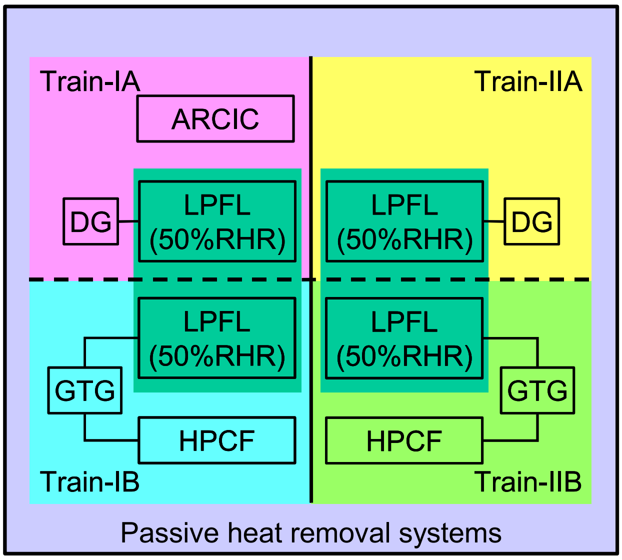


FIG. 4.2-7. ABWR-II ECCS configuration

As to emergency power sources for active components in RHR/RCW/RSW systems, a four-division configuration consisting of two diesel generators and two gas turbine generators is applied to increase diversity and to facilitate maintenance. On-line maintenance will be applied to the diesel generators. The gas turbine generators are expected to be maintenance free. Therefore, increased reliability and a reduced maintenance outage period will be achieved by this optimized division combination of two and four.

Since large break LOCA has been eliminated by adopting the RIP, LOCA is not the limiting event for ECCS capacity. Actually, high pressure injection system capacity is determined from reactor water level set point requirements during transients such as loss of feedwater, and low pressure injection system capacity is a result of optimum balance of residual heat removal system design. Utilizing these injection systems as ECCS, core covery throughout the entire LOCA spectrum is achieved. Figure 4.2-8 shows an example of reactor water level transient during typical LOCA assuming not only single failure but also on-line maintenance for one train of the low pressure injection system.

Containment design employs conventional pressure suppression as proven and cost-effective technology. Flow capacity of vent pipes and SRV discharge lines are increased from those of current ABWR reflecting increased power, and the large capacity SRV quencher design resolves layout restriction in the suppression pool. Suppression pool water inventory is determined considering heat sink capacity requirements for all design basis events (LOCA blowdown and SRV discharge during reactor isolation event).

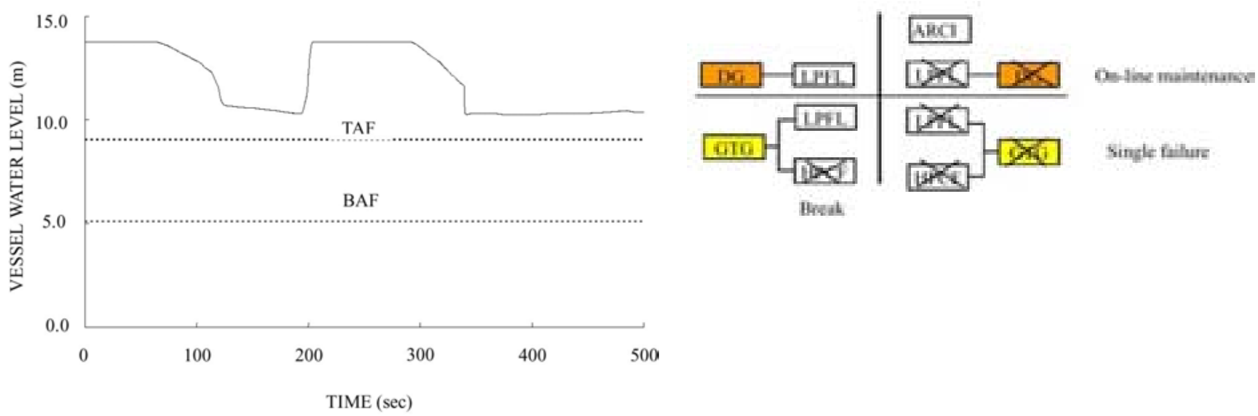


FIG. 4.2-8. The result of the DBA LOCA analysis by SAFER code

4.2.6.3 *Severe accidents (beyond design basis accidents)*

ABWR-II ECCS network has in-depth capability of redundant high pressure injection similar to that of ABWR, with extended capability. The advanced reactor core isolation cooling (ARCIC) system has capability of self-standing operation and power supply under long-term station blackout (SBO) condition beyond battery capacity. In-depth inventory makeup is performed by HPCF as a backup of ARCIC for loss of feedwater event. In the event that emergency operating procedure is called, any single ECCS pump can maintain fuel cladding temperature and oxidation below PSA success criteria (1200 ^{O}C and 15 %) utilizing depressurization system as needed.

One of the new features of ABWR-II safety design is adoption of passive systems. The passive heat removal system (PHRS) consists of two dedicated systems, namely passive reactor cooling system (PRCS) and passive containment cooling system (PCCS), and common heat sink pool above the containment allowing one day grace period (Figure 4.2-9). These passive systems not only cover beyond DBA condition, but also provide in-depth heat removal backup for RHR, and practically eliminate necessity of containment venting before and after core damage as a means of overpressure protection.

Flammable gas control in the containment is performed by the combination of inerting and passive autocatalytic recombiner (PAR) which has advantages both of safety (automatic startup and passive operation) and economy (low cost, flexible layout and easy maintenance).

The containment design considers severe accident phenomena such as direct containment heating (DCH), fuel coolant interaction (FCI), and molten core concrete interaction (MCCI) on a safety margin basis. The Japanese industry, collaborating with experts in research organizations, has recently established guidelines for containment performance design/evaluation under severe accident, and detailed quantitative examination from both phenomenological and probabilistic aspects is underway.

Preliminary PSA evaluation shows that core damage frequency (CDF) for internal events during power operation has been reduced about one third of ABWR (See Figure 4.2-10) as a result of emergency power diversity and redundancy enhancement, passive cooling system installation, and self-standing ARCIC.

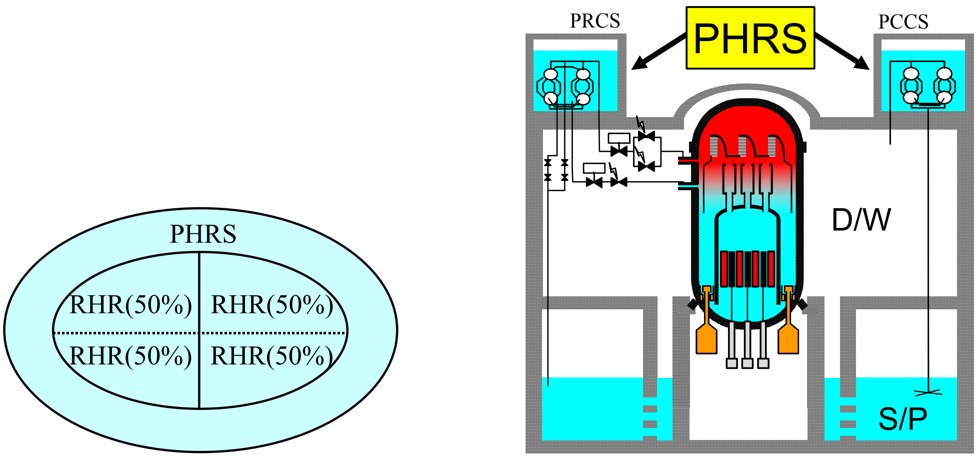
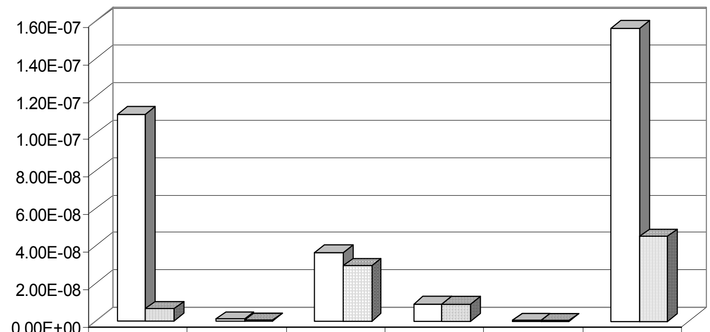


FIG. 4.2-9. Passive heat removal system



	SBO	LOCA	Transients	LOOP	ATWS	Total CDF
SSAR	1.10E-07	6.80E-10	3.66E-08	9.04E-09	2.70E-10	1.57E-07
ABWR II	6.19E-09	2.68E-10	2.94E-08	9.10E-09	2.19E-10	4.52E-08

FIG. 4.2-10. Results of level 1 PSA for the ABWR-II and the ABWR

4.2.7 Plant layout

4.2.7.1 Buildings and structures, including plot plan

No information provided.

4.2.7.2 Reactor building

Reactor building layout which incorporates the following current reference design concepts was studied:

- Rationalized 4 division RHR
- Passive heat removal systems
- RCIC with a generator
- Rationalized CR/CRD by function
- Gas turbine generators in addition to diesel generators

Since the reactor building layout for the current ABWR is highly optimized, the reference ABWR-II reactor building layout was modified in order to accommodate the above features and to reduce construction costs.

The ABWR-II reactor building is a steel plate reinforced concrete structure. The integrated reactor building and cylindrical containment structure is adopted to improve constructability, to reduce construction costs and the construction schedule. A secondary containment surrounds the primary containment and provides a second containment function including a standby gas treatment system.

Key features of the ABWR-II reactor building layout include:

(a) Suppress the increment of building volume due to power up-grade from current ABWR,
(b) Keep the same operability and maintainability as current ABWR,
(c) Keep simple structural shapes to improve constructability (reduce costs and schedule).

Two types of reactor building layout complying with slightly different Primary Containment Vessel (PCV) configurations were studied. The volume of these ABWR-II reactor buildings were approximately 102%-104% of the current ABWR (18%-20% less than the current ABWR at per power ratio) while keeping the advantages of the current ABWR such as operability and maintainability, etc.

4.2.7.3 Containment

The primary containment vessel configuration which incorporates the following current reference design concepts was studied:

- Large capacity SRV
- Low pressure drop MSIV

Two types of PCV configurations, the Modified ABWR containment and the Separated Drywell containment, were studied.

(1) Modified ABWR containment

The basic configuration is based on the reinforced concrete containment vessel of the proven ABWR:

This containment has the following features:

(a) Reinforced concrete will provide the strength necessary to withstand the pressure, and an interior steel liner will ensure the required air tightness;
(b) Cylindrical structure is integrated with reactor building;
(c) The top slab serves as a portion of the spent fuel pool and dryer-separator pool;
(d) Vessel features a horizontal vent, access tunnel in the suppression chamber and rigid diaphragm floor.

Reflecting the consideration of the thermal output level, the diameter and height of the containment vessel were carefully reviewed from a safety-design point of view, so the height of the containment has been increased. (See Figure 4.2-11)

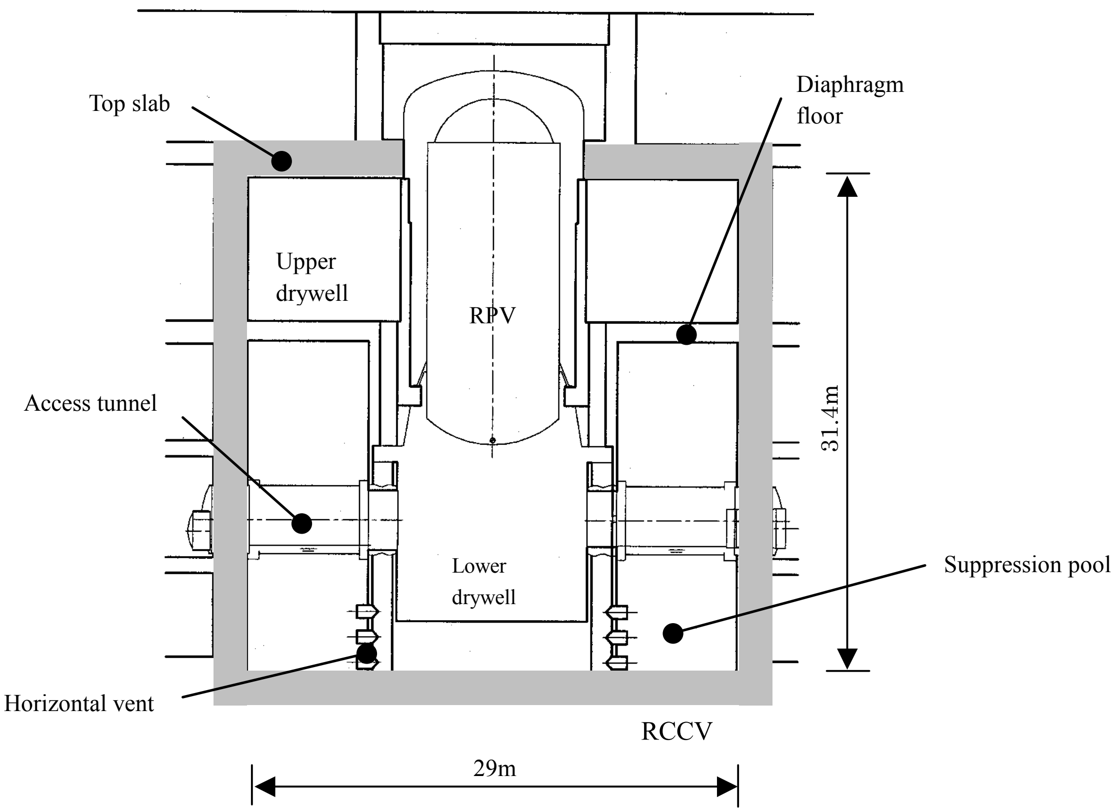


FIG. 4.2-11. Section View of Modified ABWR PCV

(2) Separated Drywell containment

Separated Drywell containment, which is being studied as a candidate for ABWR-II primary containment, has the following features:

(a) The basic configuration and functional capability are basically the same as conventional BWR containment which comprises the drywell (DW), wetwell (WW) and supporting system based on pressure suppression type. The containment structure is designed to maintain its functional integrity against the transient pressure and temperature which occur following any postulated loss of coolant accident (LOCA).

(b) The most characteristic feature is separation of drywell, i.e., the drywell is separated at RPV skirt into upper drywell (UD) and lower drywell (LD). Each drywell zone has its own vent pipes to wetwell. Vacuum breakers are installed between upper drywell and wetwell, and between lower drywell and wetwell, respectively. (See Figure 4.2-12)

(c) If a pipe break occurs in upper drywell, steam flux flows to wetwell through vent pipes, then vacuum breakers between wetwell and lower drywell open, thus lower drywell works as portion of wetwell air space during and after LOCA.

(d) These features make it easy to keep enough wetwell air space. In other words, separated drywell containment has the capability of reducing pressure during not only LOCA but also severe accident without enlarging PCV volume or venting excessive hydrogen to atmosphere, respectively. Figure 4.2-13 shows the section view of the drywell separated PCV.

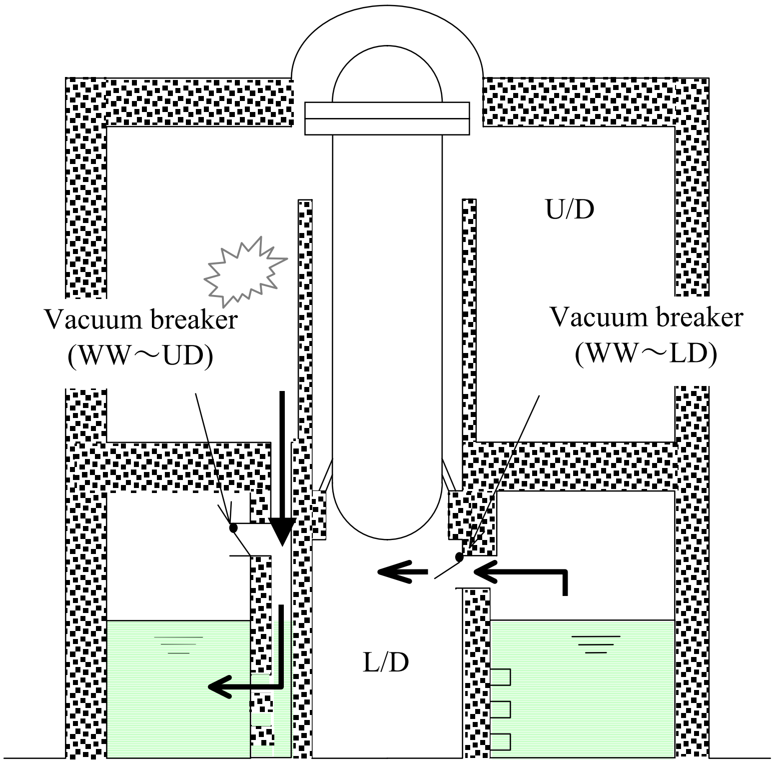


FIG. 4.2-12. Basic Principle of Drywell Separated PCV

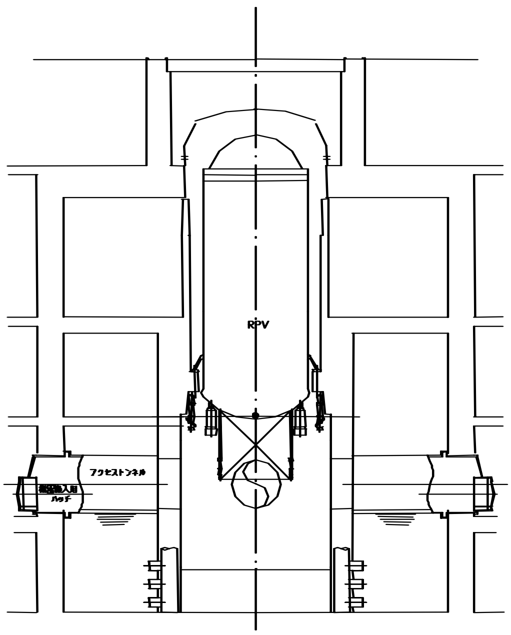


FIG. 4.2-13. Section View of Drywell Separated PCV

4.2.7.4 Turbine building

No information provided.

4.2.7.5 Other buildings

No information provided.

4.2.8 Technical data

General plant data

Power plant output, gross	1717	MWe
Power plant output, net	1638	MWe
Reactor thermal output	4960	MWth
Power plant efficiency, net	33.0	%
Cooling water temperature	≈ 28.0	°C

Nuclear steam supply system

Number of coolant loops	1	
Primary circuit volume		m^3
Steam flow rate at nominal conditions	2681	kg/s
Feedwater flow rate at nominal conditions	2677	kg/s

Reactor coolant system

Primary coolant flow rate	15667	kg/s
Reactor operating pressure	7.17	MPa
Steam temperature/pressure	287.8/7.17	°C /MPa
Feedwater temperature	215.5	°C
Core coolant inlet temperature	277	°C
Core coolant outlet temperature	288	°C
Mean temperature rise across core	11	°C

Reactor core

Active core height	3.71	m
Equivalent core diameter	5.41	m
Heat transfer surface in the core		m^2
Average linear heat rate		kW/m
Fuel weight	190	tU
Average fuel power density	26.1	kW/kg U
Average core power density	58.1	kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		
Fuel material	Sintered UO_2	
Fuel (assembly) rod total length		mm
Rod array	(8×8)×4,square lattice	
Number of fuel assemblies	424	
Number of fuel rods/assembly	56×4	
Number of spacers		
Enrichment (range) of first core, average		Wt%
Enrichment of reload fuel at equilibrium core	5.2	Wt%
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel [capability]		>60,000 MWd/t
Cladding tube material	annealed, recrystallised Zr 2	
Cladding tube wall thickness		
Outer diameter of fuel rods	10.3	mm
Fuel channel/box; material	Zr-4 or Zr-2	
Overall weight of assembly, including box		kg
Uranium weight/assembly	448	kg
Active length of fuel rods	3.71×10^3	mm
Burnable absorber, strategy/material	axial and radial grading/ Gd_2o_3 mixed with fuel	
Number of control rods	197	
Absorber material	B_4C and Hafnium	
Drive mechanism	electro-mechanical/hydraulic	
Positioning rate	<33	mm/s
Soluble neutron absorber	Boron	

Reactor pressure vessel

Inner diameter of cylindrical shell	7450	mm
Wall thickness of cylindrical shell	190	mm
Total height, inside	21300	mm
Base material; cylindrical shell	low-alloy carbon steel	
RPV head		
lining	stainless steel	
Design pressure/temperature	8.62/302	MPa/°C

Transport weight (lower part w/rigging)		t
RPV head		t

Reactor recirculation pump

Type variable speed, wet motor, single stage, vertical internal pump

Number	10	
Design pressure / temperature	same as for RPV	MPa/°C
Design mass flow rate (at operating conditions)		1725 (each)
	kg/s	
Pump head	0.347	MPa
Rated power of pump motor (nominal flow rate)		kW
Pump casing material	same as for RPV	
Pump speed (at rated conditions)	≤ 1500	rpm
Pump inertia		kg m^2

Primary containment

Type	Pressure-suppression/reinforced concrete	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)		
Modified ABWR	29/31.4	m
Separated Drywell	29/31.2	m
Design pressure/temperature	310/171	kPa/°C
Design leakage rate		vol%/day
Is secondary containment provided?		Yes

Reactor auxiliary systems

Reactor water cleanup, capacity	53.6	kg/s
filter type	deep bed	
Residual heat removal, at high pressure		kg/s
at low pressure (100□)	239.6×4	kg/s
Coolant injection, at high pressure (HPCF)	36.1×2	kg/s
(ARCIC)	69.4×1	kg/s
at low pressure (LPFL)	239.6×4	kg/s

Power supply systems

Main transformer,		
rated voltage		kV
rated capacity		MVA
Plant transformers,		
rated voltage (secondary)	/11.5-6.9	kV
rated capacity	130/80-50	MVA
Start-up transformer		
rated voltage (secondary)	/11.5-6.9	kV
rated capacity	80/40-40	MVA
Medium voltage busbars (6 kV or 10 kV)	11.5/6.9	kV
Number of low voltage busbar systems		
Standby diesel generating units; number	2	MW
rated power		
Standby gas turbine generating units;		
Number	2	MW
rated power		
Number of diesel-backed busbar systems	2	
Voltage level of these	6.9	kV AC
Number of gas turbine-backed busbar systems		2
Voltage level of these	6.9	kV AC
Number of DC distributions	4	
Voltage level of these	125	V DC
Number of battery-backed busbar systems	4	
Voltage level of these	125	V DC

Turbine plant

Number of turbines per reactor	1	
Type of turbine (s)	six flow, tandem compound single reheat	
Number of turbine sections per unit (e.g. HP/LP/LP)		1 HP / 3 LP
Turbine speed		rpm
Overall length of turbine unit		m

Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/°C

Generator

Type	3-phase, turbo-generator	
Rated power		MVA
Active power	1700	MW
Voltage		kV
Frequency		Hz
Total generator mass, including exciter		t
Overall length of generator		m

Condenser

Type	shell type (3 shells)	
Number of tubes	1 tube pass/shell	
Heat transfer area		m^2
Cooling water flow rate		m^3/s
Cooling water temperature		°C
Condenser pressure (HP shell)		kPa

Condensate pumps

Number	
Flow rate	kg/s
Pump head	MPa
Temperature	°C
Pump speed	rpm

Condensate clean-up system

Full flow/part flow
Filter type (deep bed or rod type)

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/°C

Feedwater pumps

Number	
Flow rate	kg/s
Pump head	Mpa
Feed pump power	MW
Feedwater temperature (final)	°C
Pump speed	rpm

Condensate and feedwater heaters

Number of heating stages, low pressure
high pressure
feedwater tank

4.2.9 Measures for improving economy and maintainability

In planning for a future reactor, it is indispensable to set a cost target of power generation. It has become more and more difficult for nuclear power plants to keep cost competitiveness over other forms of power. For ABWR-II as a future plant of the late 2010's, the challenging target of 30 % reduction in power generation cost from that of a standardized ABWR was set. Nuclear power plants have relatively high construction cost and low running cost compared to fossil power plants. Therefore, capital cost reduction by design has been carefully looked into in addition to operation and maintenance (O&M) cost reduction.

The following are design considerations to improve ABWR-II economics.

4.2.9.1 Design simplification

When the Phase I programme started, the ABWR-II plant power output was set at 1350 MWe, the same as ABWR. During Phase II, when the need for cost reduction increased, the reference output was increased to 1500 MWe to obtain larger merit by economies of scale. In Phase III, it became apparent the target of 20 % power generation cost reduction was so challenging that further output increase should be required. The output was again increased to 1700 MWe as a reference.

This 1700 MWe output was decided considering compatibility with Japanese grid capacity and manufacturability for components such as reactor pressure vessels and generators. The larger output would be suitable also for future replacement of old plants because of better efficiency in using limited site area and common facilities.

The new type MSIV for ABWR-II is now under development. The bore diameter of the MSIV is increased and the center of gravity of its driving mechanism is lowered. This diameter increase is not simply an enlargement from ABWR but is optimized in such a way that the pressure loss will be decreased from ABWR to increase efficiency. The lowered center of gravity of the driving mechanism by relocating springs and oil damper will contribute to improvement of seismic capability. This design makes it possible to simplify supporting rigs against seismic.

For the ABWR-II, an increase of discharge capacity and simplification of valve structure were considered. In order to decrease the number of SRVs, the discharge capacity per SRV is increased by 70% to 680 t/h (nominal) from ABWR's 395 t/h with an increased throat diameter and increased coil spring diameter. At the same time, the structure of SRV is simplified by integrating an air cylinder into the SRV's main body.

In order to have enough flexibility for future higher burnup and longer cycle operation and to reduce components and shorten the refueling outage time, a 1.5 times larger fuel bundle and K-lattice were selected for the reference design. Though power output of ABWR-II is increased to 1700MWe from 1356MWe, the number of fuel bundles is decreased to 424 from 872, and the number of control rods is decreased to 197 from 205.

In the CRD design of ABWR-II, an improvement in the connection between motor and CRD shaft was made. The improved mechanism is called a magnetic coupling. The magnetic coupling can transmit torque between the CRD motor and the CRD shaft through the pressure boundary without a penetration instead of having a seal around a penetrating shaft. This improvement eliminates the sealing parts where inspection and maintenance are most necessary. In addition, the improved HCU for CRD, which is under development, will be adopted to ABWR-II. In case of conventional HCUs, 99 HCUs are required for 197 CRs. By applying the improved ones, 65 improved HCUs and one conventional HCU are enough for them.

Optimization was made in the RHR system together with the RCW and RSW systems. Taking into consideration that the passive heat removal systems of ABWR-II can be considered as backup, the basic configuration of RCW is two divisions instead of the three in ABWR. This two-division configuration is expected to reduce equipment cost for RCW that has relatively large amount of materials especially for piping.

Flammable gas control in the containment is performed by the combination of inerting and passive autocatalytic recombiner (PAR) which has advantages both of safety (automatic startup and passive operation) and economy (low cost, flexible layout and easy maintenance).

The ABWR-II reactor building is a steel plate reinforced concrete structure. The integrated reactor building and cylindrical containment structure is adopted to improve constructability, to reduce construction costs and the construction period. Two types of reactor building layout complying with slightly different Primary Containment Vessel (PCV) configurations were studied. The volume of these ABWR-II reactor building were approximately 102%-104% relative to the current ABWR (meaning 18%-20% less than the current ABWR at per power ratio) while keeping the advantages of the current ABWR such as operability and maintainability, etc.

ABWR-II plant system features are summarized in Figure 4.2-14.

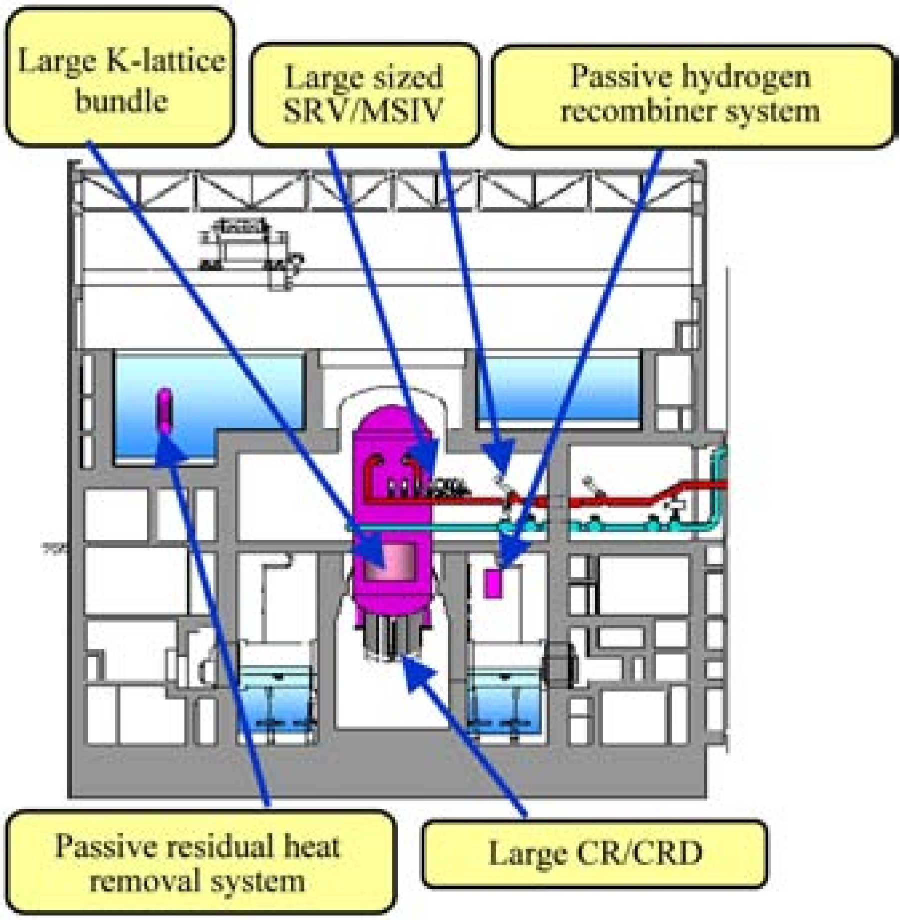


FIG. 4.2-14. Features of ABWR-II Plant System

4.2.9.2 Operation flexibility improvement

Targeted operating cycle length and average discharge burnup of the ABWR-II core are 18 effective full power months (EFPM) and 60GWd/t, respectively. The recirculation flow control range of 20% is achieved. In the BWR, the excess reactivity is controlled by burnable poison, CRs and recirculation flow. The recirculation flow can control the reactivity through the void fraction, which changes the neutron spectrum in the core. In order to reduce use of CRs under ordinary operation and improve the neutron economy, it is desirable to enlarge the controllable reactivity using recirculation flow. The reference core design has a more negative void coefficient due to larger fuel inventory, and reactivity control capability with core flow becomes larger than current core design. In order to enlarge the reactivity change due to unit recirculation flow change further, the SSR has been developed as an option. to be used instead of the water rod, in which the water level develops naturally during operation and changes according to the recirculation flow rate through the channel. In this case, it is expected that fuel cycle cost will be reduced and that CRs need not be replaced periodically, contributing to additional refueling time reduction.

For RHR, RSW and active components in RCW, in total, make up a four-division configuration that facilitates on-line maintenance and increases reliability and safety. As to emergency power sources for active components in RHR/RCW/RSW systems, a four-division configuration consisting of two diesel generators and two gas turbine generators is applied to increase diversity and to facilitate maintenance. On-line maintenance will be applied to the diesel generators. The gas turbine generators are expected to be maintenance free. Therefore, increased reliability and a reduced maintenance outage period will be achieved by this optimized division combination of two and four.

A new feature of ABWR-II design is adoption of passive safety systems. These systems not only cover beyond DBA condition, but also provide in-depth heat removal backup for RHR, and practically eliminate necessity of containment venting before and after core damage as a means of overpressure protection. Flammable gas control in the containment is performed by the combination of inerting and passive autocatalytic recombiner (PAR) which has advantages both of safety (automatic startup and passive operation) and economy (low cost, flexible layout and easy maintenance).

4.2.9.3 Cost reduction of equipment and structures

Cost reduction of ABWR is pursued in various equipments. Improving equipment itself and/or decreasing number of equipments attain cost reductions of equipment. In additions, improvement of plant efficiency contributed to cost reduction is attained by improvement of equipment such as pressure drop reduction. Major cost reduction equipments are shown in followings.

- Low pressure drop type separator (Refer to Ref. [1])
- Large capacity and simplified mechanism SRV
- Low pressure drop MSIV improving plant efficiency
- Simplified mechanism CRD by magnet coupling
- Horizontal type heat exchanger for PCCS

The ABWR-II reactor building is a steel plate reinforced concrete structure. The integrated reactor building and cylindrical containment structure is adopted to improve constructability, to reduce construction costs and the construction schedule. A secondary containment surrounds the primary containment and provides a second containment function including a standby gas treatment system.

4.2.9.4 Reduction of construction period

ABWR achieved construction schedule in 37 months (from first concrete work to fuel loading) at Kashiwazaki-Kariwa FOAKE plant. Construction schedule target of ABWR-II is 29.5 months. This remarkable reduction of construction schedule is achieved by the following procedures including modified ABWR construction technologies:

- Large scale modularization
- Open-top installation method
- Rationalization of testing
- Integrated module of both mechanical and civil

- SC structure building
- Extra large scale crane
- Expansion of the application of automatic welding machinery

4.2.9.5 Scope reduction of the maintenance during operation and outages

The large 1.5 K-lattice core arrangement enables to decrease numbers of fuel bundles and CRDs. These contribute to additional refueling time reduction and CRDs maintenance time reduction.

The magnetic coupling can transmit torque between the CRD motor and the CRD shaft through the pressure boundary without a penetration instead of having a seal around a penetrating shaft. This improvement eliminates the sealing parts where inspection and maintenance are most necessary, and also the seal detection system that requires monitoring during operation, thus making CRDs maintenance free, contributory to maintenance cost reduction.

For RHR, RSW and active components in RCW, in total, make up a four-division configuration that facilitates on-line maintenance and increases reliability and safety. As to emergency power sources for active components in RHR/RCW/RSW systems, a four-division configuration consisting of two diesel generators and two gas turbine generators is applied to increase diversity and to facilitate maintenance. On-line maintenance will be applied to the diesel generators. The gas turbine generators are expected to be maintenance free. Therefore, increased reliability and a reduced maintenance outage period will be achieved by this optimized division combination of two and four. This on-line maintenance changes the scope of the maintenance during operation and outages.

Passive system such as PCCS and PAR are applied for ABWR-II. Passive system needs not to do maintenance work as active system. Therefore, scope reduction of the maintenance during operation and outages is achieved against passive related system.

4.2.9.6 Making the maintenance easier and with lower radiation exposure

The large 1.5 K-lattice core arrangement enables to decrease numbers of fuel bundles and CRDs. These contribute to additional refueling time reduction and CRDs maintenance time reduction. Therefore the large 1.5 K-lattice core arrangement make the maintenance easier and with lower radiation exposure.

Simplified mechanism and large capacity type SRV is applied for ABWR-II. This SRV contribute to make the maintenance easier and with lower radiation exposure.

Magnet coupling CRD eliminates the sealing parts where inspection and maintenance are most necessary, and also the seal detection system that requires monitoring during operation, thus making CRDs maintenance free, contributory to maintenance easier and with lower radiation exposure.

Taking into consideration that the passive heat removal systems of ABWR-II can be countered as backup, the basic configuration of RCW is two divisions instead of the three in ABWR. This two-division configuration is expected to maintenance easier and with lower radiation exposure.

4.2.9.7 Increasing the capacity factor

Refueling outage period reduction is a major factor in power generation cost reduction because it improves the capacity factor. In the ABWR-II programme, outage period reduction has been approached from two aspects:

- Design for maintenance reduction or on-line maintenance
- Expected future deregulation

Examples of design for maintenance reduction or on-line maintenance are:

- Reduced number of fuel bundles
- CRD boundary penetration shafts elimination
- Reduced number and simplification of SRVs
- Four-division RSW system configuration
- Four-division emergency power sources

As to deregulation, maintenance interval extension and rationalization of regulatory audit schedule and test items was taken into consideration.

After having checked feasibility of a 30-day refueling outage period, the current target is a further reduction to 20 days. Considering an operation cycle of 18 months, the capacity factor with a 20-day refueling outage will be 96 %. Since this 20-day refueling period considers minimum maintenance work, there will be some longer outages once in a while through the plant lifetime. The average capacity factor through the plant lifetime would be expected to be more than 90 %.

4.2.9.8 Reduction of the power generating cost

There are several ways to reduce power generation cost for new nuclear power plants. Figure 4.2-15 shows the breakdown of the nuclear power generation costs as reported by the Agency of Natural Resources and Energy in Japan. The power generation cost is largely divided into three categories: capital cost, operation and maintenance (O & M) costs and fuel cost. Since the capital cost accounts for 39% of the total, its reduction would seem to be the most effective way. However, simple elimination of components is not a good choice. For example, engineering safety features are around 6% of the total cost, thus the impact on total cost reduction is rather small even if they could be eliminated completely. Therefore, increasing plant output while minimizing impact on the plant systems is the best way to reduce the capital cost. Moreover, O & M costs are almost constant regardless of the plant output. In short, two dominant factors, which occupy 71% of the power generation costs, can be directly reduced by the plant output increase.

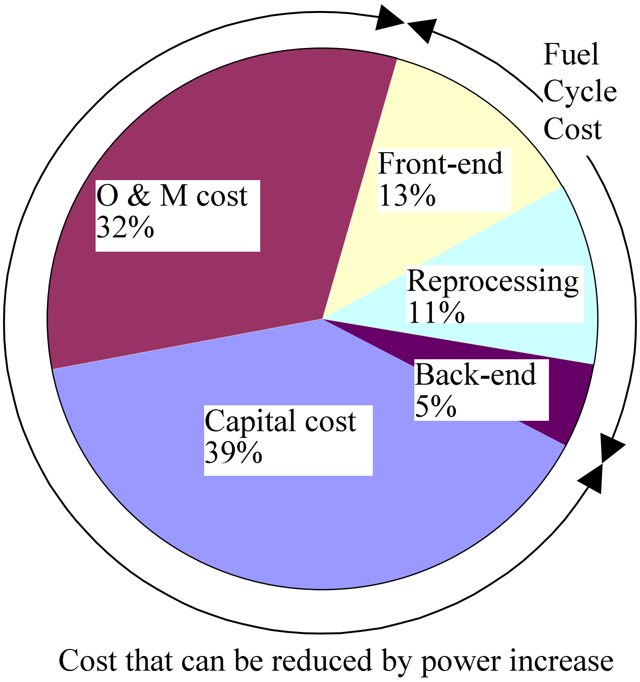


FIG. 4.2-15. Power Generation Cost Breakdown

Figure 4.2-16 summarizes cost reduction effects on the power generation costs by introducing the ABWR-II. Based on the current reference design concept as shown above, the plant construction cost for a 1700 MWe ABWR-II is estimated to be almost the same as that of a 1350 MWe ABWR. Therefore, the 26% power increase from 1350MWe to 1700MWe directly affects the capital cost and is expected to reduce it by approximately 20%. Design simplification, operation flexibility improvement and increasing the capacity factor make it possible to reduce the O&M costs remarkably. About 20% reduction is also expected for O&M costs. Burn-up extension up to 60GWd/t by incorporating the SSR allows fuel cycle cost to be reduced by 7%.

Overall cost reduction for power generation is estimated to be about 15-20% against the ABWR.

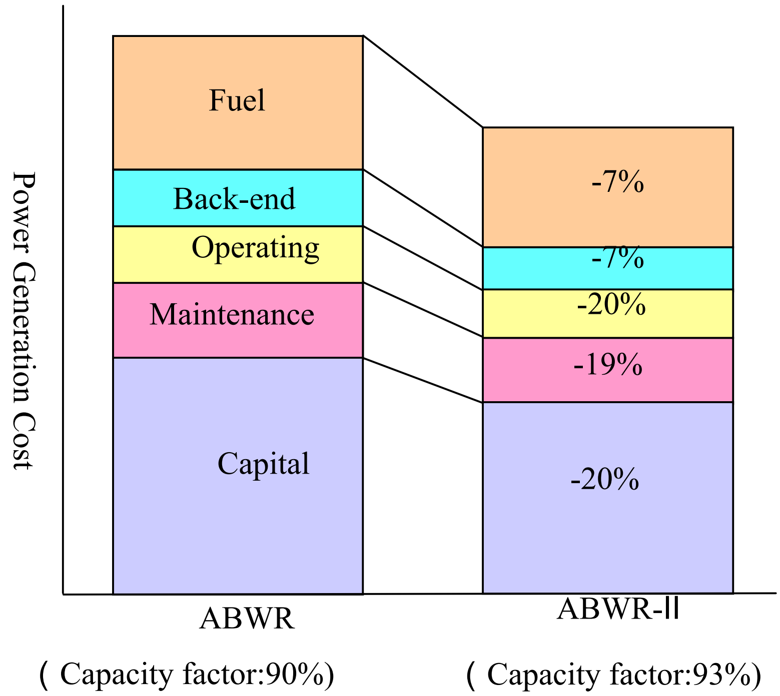
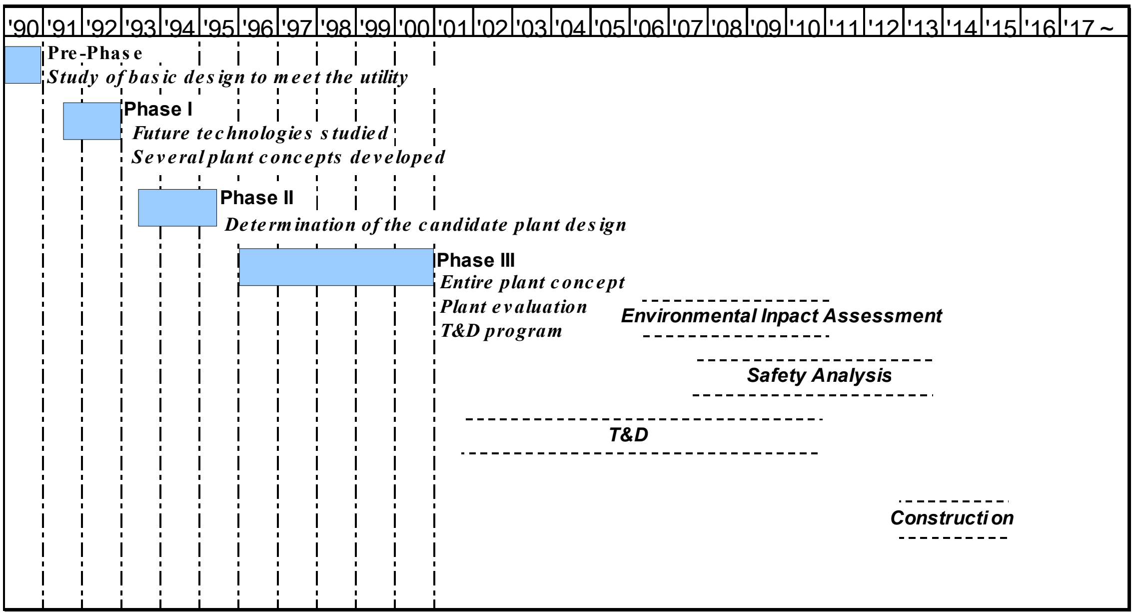


FIG. 4.2-16. Cost Reduction Effect of ABWR-II

4.2.10 Project status and planned schedule

The ABWR-II development project was initiated over a decade ago and has completed three phases to date. In Phase I (1991-92), basic design requirements were discussed and several plant concepts were studied. In Phase II (1993-95), key design features were selected in order to establish a reference reactor concept. In Phase III (1996-2000), based on the reference reactor concept, modifications and improvements were made to fulfill the design requirements. And in the present, various testing programmes are performed or planned to consolidate their feasibilities and to find further room for improvements. The commercial introduction of ABWR-II is now set by the latter half of the 2010s when replacements of old nuclear power plants are expected to start.



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4.3 APWR (MITSUBISHI, JAPAN/WESTINGHOUSE, USA)

4.3.1 Introduction

Nuclear power generated by light water reactors accounts for approximately 1/3 of Japan's power supply. Also, it is expected to play an important role in providing energy security and preservation of the global environment in the future.

The advanced PWR (APWR) has been developed, as a nuclear power plant for future use in Japan, as a joint international cooperative development project by seven companies comprising the five PWR electric power companies (Hokkaido, Kansai, Shikoku, Kyushu Electric Power Company, and Japan Atomic Power Company) and Mitsubishi Heavy Industries and Westinghouse. Its development was part of Phase III of the Improvement and Standardization Program of the Ministry of International Trade and Industry (Currently the Ministry of Economy, Trade, and Industry). In the APWR, advanced technologies based on the operational experience gained so far have been incorporated. Also safety, reliability, operability and the performance of the plant have been further increased, and the construction cost has been further reduced due to the benefit of economy of scale resulting from the increase in capacity. The first APWR plant will be adopted by the Japan Atomic Power Company, Tsuruga-3 and 4. Here we introduce some outstanding features of this new APWR.

4.3.2 Description of the nuclear systems

4.3.2.1 Primary circuit and its main characteristics

Table 4.3-I shows a comparison of some major parameters between the APWR and an existing four-loop plant. The APWR is in the largest capacity class of LWRs in Japan and has adopted, for example, high performance steam generators and low pressure turbines with approx. 54 in. (1375 mm) last stage blades. Various improvements have been incorporated in the reactor core so that operation with long fuel cycles is possible using low enriched fuel in order to reduce uranium requirements, and to provide increased flexibility for various application such as the use of MOX cores and high burn-up fuels.

Also critical equipment such as reactor internals and steam generators have been designed taking into account operational experience of aging on operating plants so that a high degree of reliability can be obtained. To ensure safety, the reliability of the equipment and systems has been increased, and highly advanced safety systems such as methods of providing assistance to the operations during abnormal events have been adopted. 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, and also the latest electronics technologies to improve the man machine interface have been introduced in the main control room. In addition, in order to make the plant easier to maintain, a variety of improved technologies have been inconporated, thus improving the efficiency of periodical inspections and reducing exposure to employees.

4.3.2.2 Reactor core and fuel design

The reactor core, consisting of 257 17 × 17 fuel assemblies, has a core thermal output of approx. 4,451 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. Also the core has 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 of long operating cycles.

TABLE 4.3-I. MAJOR APWR PARAMETERS

	APWR	Existing four loop PWR
Electric power output	1,538 MWe	1,180 MWe
Core thermal output	4,451 MWt	3,411 MWt
Reactor thermal output	4,466 MWt	3,423 MWt
Fuel type	17 × 17	17 × 17
Number of fuel assemblies	257	193
Fuel effective length	Approx. 3.7 m	Approx. 3.7 m
Total uranium inventory	Approx. 121 tonnes U	Approx. 89 tonnes U
Number of control rods	69	53
Neutron reflector	Stainless steel	None
Reactor vessel	Approx. 5.2 m inner dia. & Approx. 14 m height	Approx. 4.4 m inner dia. & Approx. 13 m height
Steam generators	70F-1 type	52F type
Primary coolant pumps	100A type	93A-1 type
Primary system flow (m^3/h/loop)	Approx. 25,800	Approx. 20,100
Turbine	TC6F54	TC6F44
Generator	1,715 MVA	1,310 MVA
Containment	PCCV	PCCV
Engineered safety systems	Four trains of mechanical systems & Two trains of power systems	Two trains
Refuelling water storage pit	Inside containment	Outside containment
Reactor protection system	Digital	Analog
Reactor control system	Digital	Digital
Main control room	Improved	Standard

To reduce fuel cycle costs, the fuel assemblies have zircalloy grids with low neutron absorption and the core is surrounded with a reflector to reduce neutron leakage, thus increasing neutron efficiency.

The reactor uses the 17 × 17 fuel which has operated well in existing plants. The design is made by adopting zircalloy grids with low neutron absorption as above-mentioned and other means, in order that it can be used for high burn-ups and increased loadings of MOX fuel.

4.3.2.3 Fuel handling and transfer systems

The fuel handling and fuel transfer systems cousist mainly of the refuelling crane, fuel transfer system, and multi-functional mast type spent fuel pit crane as in existing plants.

Considering the recent need to reduce the periodical inspection time, many improvements including an increase in the speed of each system have been introduced.

Also, in order to reduce operations in radiation controlled areas, these systems can be operated automatically from a remote centralized control room instead of the present method of operating individually from a control station next to each piece of equipment.

4.3.2.4 Primary components

Reactor pressure vessel

Although the reactor vessel inside diameter has been increased to 5.2 m in order to accomodate 257 fuel assemblies, the vessel is made with forged rings and has no longitudinal welds in the core area as is the case with the latest four-loop plant. The neutron irradiation of the steel opposite the core has been reduced to approx. 1/3 as compared with the 4-loop plant reactor vessel with a 40-year neutron fluence of about 2×10^{19} n/cm^2 by providing a neutron reflector, thus increasing the reliability of the reactor vessel.

Also, in order to reduce the 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 top dome of the reactor vessel is designed to be reduced to reactor inlet temperature.

A comparison between the APWR and an existing four-loop plant is shown in Figure 4.3-1.

Reactor internals

The reactor internals are increased in the radial dimensions of the members according to an increase in size of the core. The neutron 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. By installing the neutron reflector, the neutron irradiation of the reactor vessel can be reduced to approx. 1/3 compared to present reactors. On present reactors, the core baffle is a plate structure held together with 2000 or more bolts, whereas the new neutron reflector has a simple construction which does not use bolts in the effective core area.

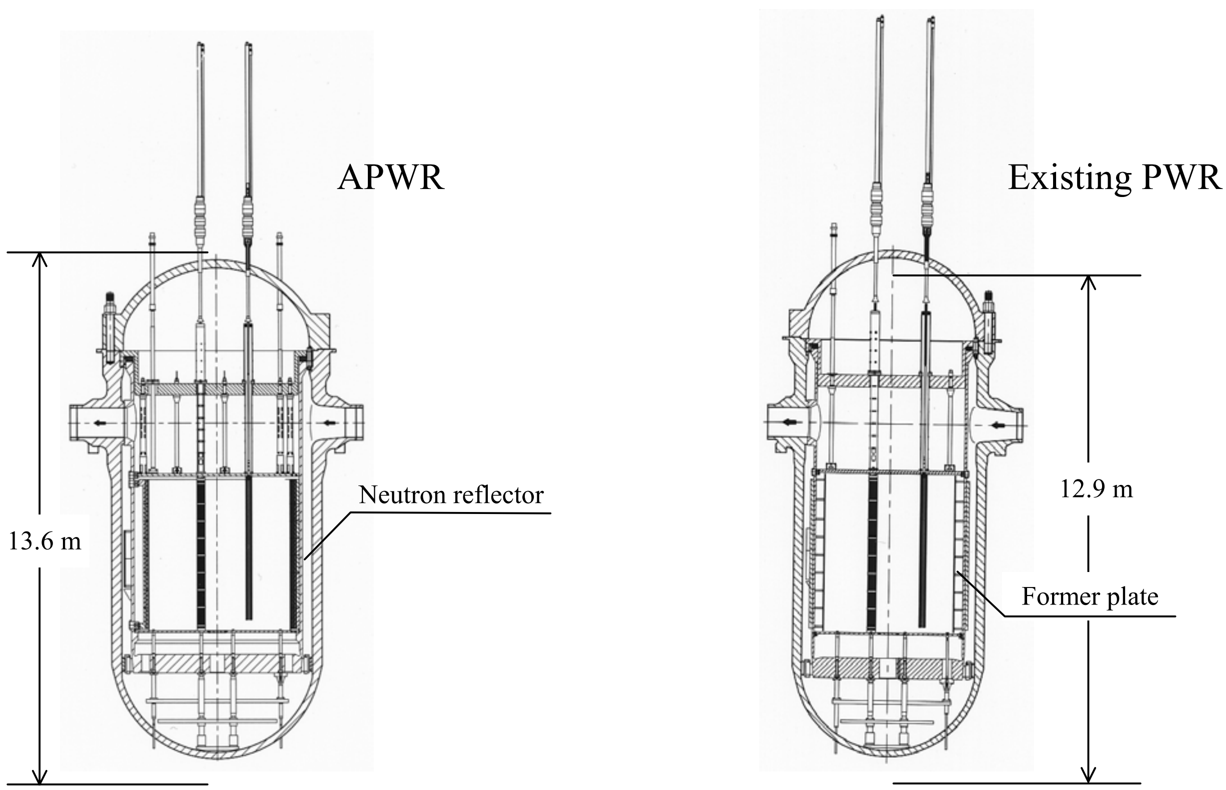

FIG. 4.3-1. A comparison between the APWR and an existing PWR

The improved core internals underwent flow tests to check the validity of the design.

Steam generators

The APWR has adopted steam generators (Type 70F-1) to match the increased capacity of the reactor core. The tubes are 3/4 in (19 mm) diameter which is smaller than the 7/8 in. (22 mm) used in existing plants. This results in a more compact steam generator with resistance to earthquakes, etc.

The tubes of the steam generators are made of thermally treated alloy 690 (TT690). Also the design of the anti-vibration bars in the U-bend area of tubes has been improved to reduce the risk of flow induced vibration of the tubes.

In addition, in order to make maintenance and inspections easier, accessibility has been improved by increasing the diameter of the manholes and in other ways. Figure 4.3-2 shows a schematic drawing of the steam generator.

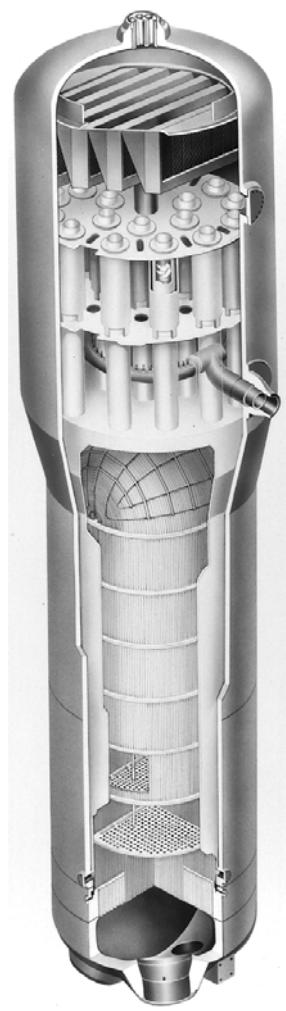

FIG. 4.3-2. APWR - Steam generator

Reactor coolant pumps

Because the primary coolant pump has to supply a flow approx. 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. For the improved No.1 seals, heat-resisting 0-rings are employed as well as ceramic material which has larger size and excellent durability, thus aiming at enhancement of the reliability.

Main coolant lines

For piping material, low alloy steel (with stainlesss steel lining) is used from the point of view of enhancement of reliability and inspectability.

4.3.2.5 Reactor auxiliary systems

Chemical and volume control system

The chemical and volume control system has the following main functions.

The first function is to adjust the amount of primary water contained in the reactor coolant system. In normal operation, the letdown and charging flows are controlled so that the water level in the pressurizer is kept at the programmed level. Seal water is injected into the primary coolant pump seals.

The second function is to adjust the concentration of boron and the quality of water contained in the primary coolant system. The concentration of boron in the primary coolant system is adjusted by adding pure water from the reactor make up system or boric acid solution as required to compensate for fuel burn up.

The quality of water in the primary system can be controlled by adding hydrazine or lithium hydroxide, passing the water through a cation demineralizer, and adding hydrogen gas to the vapor space of the volume control tank.

The third function is to purify the primary coolant. The primary coolant is purified by a demineralizer and filter in the let down line.

The let down flow is taken from the cross-over leg of the primary coolant system, and the coolant is cooled by the regenerative heat exchanger and let down heat exchanger, and then purified in the demineralizer.

To supply water to the primary coolant system and seal water to the primary coolant pumps, a charging pump is used taking water from the volume control tank.

Boric acid tank water injection via the safety injection nozzle to the core, using two charging/boron injecting pumps as well as insertion of the control rod cluster and operation of the ECCS make the core subcritical earlier at the time of an overcooling event.

4.3.2.6 Operating characteristics

The reactor is designed so that it can be operated automatically within the range of 15 to 100% of rated output by the reactor control system.

The reactor control system is designed so that it can follow the following load change 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%)
- 65% load reduction

With respect to the load fluctuation following capacity to the following electric power system, the following is provided:

- Daily load-follow operation of 100%-50%-100%
- Automatic frequency control or governor control to control system frequency over a load range of 5%.

4.3.3 Description of turbine generator plant systems

4.3.3.1 Turbine generator plant

The high pressure turbine and low pressure turbines are double flow turbines with reaction blading.

The last stage blades are approx. 54 in. (1375 mm) ISB blades (Integral Shroud Blade) to increase the electric power output and efficiency. The high performance blades, low pressure turbine casing which improves exhoust loss, etc. further increase the efficiency. Also the moisture extraction system has been improved to reduce erosion.

The 54 in. last stage blades have been subjected to vibration tests and actual load tests to demonstrate that the turbine has a high performance and reliability.

The 1715 MVA generator which is of 4-pole type, has a larger rotor diameter than that of an existing PWR plant in order to increase the output. The rotor windings are cooled internally by hydrogen gas while water is used for cooling of the stator. The excitation is provided by a brushless static system.

Figure 4.3-3 shows a view of the turbine generator unit.

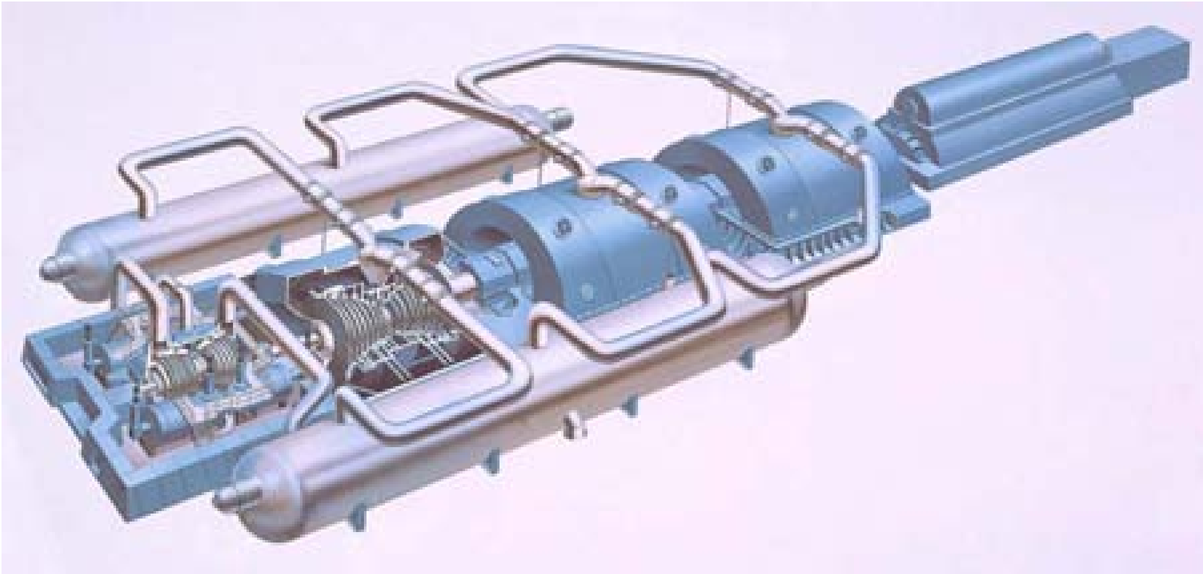

FIG. 4.3-3. APWR - Turbine generator unit

4.3.3.2 Condensate and feedwater systems

The moisture separator/reheater has a two stage heater, and can achieve a high efficiency. The turbine building has been reduced in size by reducing the outside dimensions of the moisture separator and adopting the so-called fourneck heater system. In this system four low pressure feed water heaters are installed inside of the condenser, whereas a conventional plant usually has two low pressure feed water heaters in this location.

The feed water systems use six extraction stages. These systems consist of 3 trains of four stages of low pressure heaters, the deaerator and 2 trains of single stage high pressure heaters. 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 feed water heater tubing material is changed to stainless steel. This is to avoid corrosion of the low pressure feed water heater tubes caused by ammonia and to improve water chemistry.

The capacity of the pumps installed in the condensate system is 50% × 2 units, and the capacity of the pumps installed in the feed water system is 50% x 2 units and 25% x 1 unit (backup). Even if one pump fails, partial power operation can be maintained.

4.3.3.3 Auxiliary systems

The turbine plant has the following additional features.

(a) To improve efficiency, the drains from low pressure feed water heaters are collected in the condensate system on the down stream side of the next heater.

(b) To further reduce the iron concentration in feed water, low alloy steel in the system equipment, etc. are applied.

(c) To simplify the systems and equipment, two circulating water pumps (50% capacity) have been installed in the circulating water system as is the case with the current PWRs.

4.3.4 Instrumentation and control systems

4.3.4.1 Design concept, 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.

Touch screen operations are applied, and the plant parameters and operating swiches are displayed on the same screens that are 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 among operators is improved.

4.3.4.2 Reactor protection and other safety systems

The reactor protection system and other safety systems are digital systems of the functionally distributed type.

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 distributedly in 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 and a symbolic language was used in the design. Verification and Validation tests are to be carried out to the maximum extent possible.

4.3.5 Electrical systems

4.3.5.1 Operational power supply systems

The operational AC power supply system can receive external power from the main power supply system and stand by power supply system. Power from the main power supply system comes through the main transformer and unit 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.. Therefore, the operational AC power supply system can receive power through the unit transformers continuously 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 emergency transformer which has sufficient capacity to enable the plant to be shut down safety.

The buses of the operational AC power supply system are divided into two main groups: the 6.6 kV high voltage system and the 440 V low voltage system, each comprising normal buses to supply power to loads such as primary coolant pumps, main feed water pumps and other equipment required for normal plant operation and two-trains of emergency buses to supply power to loads such as safety injection pumps and other equipment required for the safety of the plant.

In addition to the above AC power supply systems, other power systems have been provided which can be supplied from batteries in the event of an interruption or station blackout and an instrumentation and control power supply (consisting mainly of inverters) for supplying power to the instrumentation and control equipment which are mainly computer loads.

4.3.5.2 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 will start automatically immediately after an accident occurs or external power is lost, and supply power to the emergency buses. The emergency power systems are made as redundant systems, 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, and also has sufficient capacity and to supply the switchgear which must operate following a loss of external power and to supply the initial excitation power for the diesel generators. Also 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 to be consistent with the two trains and four sub-systems, and the instrumentation and control power system is divided into two trains/four channels to be consistent with the four channels.

4.3.6 Safety concept

4.3.6.1 Safety requirements and design philosophy

The configuration of the ECCS mecanical systems has been changed from the conventional two trains to four trains to give more redundancy and independence. Also, tie lines between trains have been eliminated to simplify the systems and increase the reliability.

In existing plants, refilling of the reactor vessel and reflooding of the reactor core after a LOCA are made by both the accumulators and low presssure injection pumps. In APWR, however, advanced accumulators with two-stage injection characteristics have been adopted and the present low pressure injection systems and the accumulators have been integrated to simplify the equipment and increase the functional reliability. Also, the refuelling water storage 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 needed during an accident on existing plants. A comparison between the ECCS of an existing plant and the APWR is shown in Figure 4.3-4.

As a result, it has been found that, as a whole, the APWR is expected to have a core damage frequency of at least one order of magnitude lower than the existing 4-loop plant with a core damage frequency of about 10^{-7}/ry.

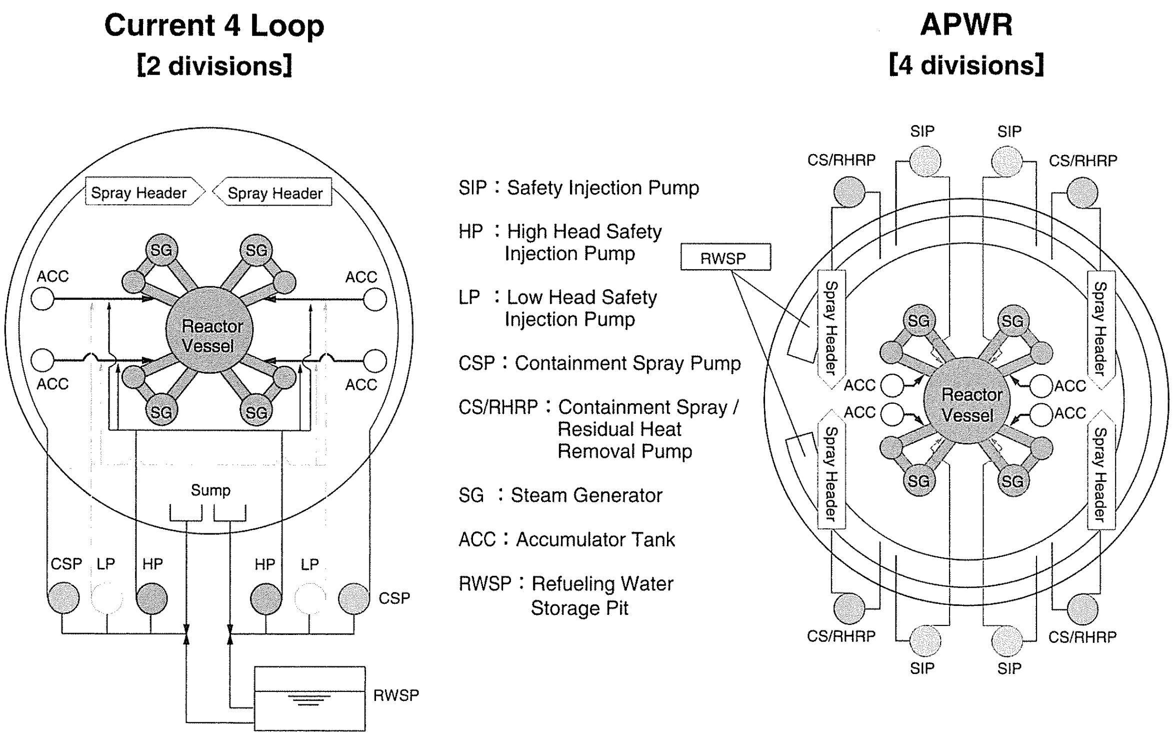

FIG. 4.3-4. APWR emergency core cooling systems compared with existing PWR

Deterministic design basis

The safety design of an APWR satisfies, from a deterministic design point of view, the safety design criteria for design basis events. Also, using probablistic assessments, the APWR is designed so that it has sufficient margins for beyond design basis events. The design basis events are abnormal operating conditions which are classified into two groups: abnormal operating transients and accidents during operation, and safety criteria have been set for each group. The standards for radiation exposure are specified for normal operation and accidents, thus reducing the risk to the general public and employees to less than an allowable limit.

Risk reduction

To further reduce the risk and provide increased protection, the reactor is designed to have a high degree of safety. Specifically, it is designed with the following design targets.

- The core damage probability 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 th of that of the latest Japanese PWR.

- The core damage frequency during shutdown should be approximately the same as the target for power operation.

- For further protection, the containment failure frequency (CFF) should be reduced to an appropriate level (to approx. 1/10 of the core damage frequency as a target).

- The structure of the containment should be designed so that its functions as a target can be maintained for one day or longer during quasi-static pressurization following a severe accident. For premature failure modes caused by missiles and dynamic loads, countermeasures should be taken for containment design, etc.

Especially, these measures are concretely classified as follows.

(a) 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 subsystem safety systems, installation of the emergency water source inside the containment, etc., countermeasures against an interfacing system LOCA and other events have been also taken which further reduce the risk.

(b) Countermeasures for increased safety during shutdowns. These include installation of an automatic interlock to isolate the letdown line when the reactor coolant system (RCS) water level is lowered, improvement of water level monitoring, improvement of the RCS water injection function during shutdowns and other countermeasures. These countermeasunes are under consideration.

(c) Countermeasures for mitigating the effects of an accident. These include the use of the containment vessel air recirculation systems, alternate sprays supplied from the fire service water systems, countermeasures for hydrogen control, etc., and, at the same time, countermeasures against the events which could become a potential threat to the containment are also studied by water injection into the cavity from the fire service systems, improvement of the cavity shape etc..

4.3.6.2 *Safety systems and features (active, passive and inherent)*

Safety systems configuration

The emergency core cooling systems and containment spray/residual heat removal 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 storage pit installed inside the containmen;
- Four advanced accumulator tanks. (Capacity: About 90 m^3/unit)

The advanced accumulators refill the reactor vessel lower plenum and downcomer immediately after a LOCA with a medium to large break size and, after that, they inject water to reflood the core and function as both the accumulator tank and low pressure injection pump of existing plants.

The safety injection pump has a function to take water from the refuelling water storage pit installed at the bottom of the containment and feed cooling water to the reactor vessel for a long period after the accident.

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 required by the steam generator when the normal feedwater system is not available. Except for the auxiliary feed water storage pit, this system consists of two mechanical sub-systems. Each sub-system consists of a motor driven auxiliary feedwater pump and a turbine driven auxiliary feedwater pump.

Emergency core cooling systems

The emergency core cooling system feeds sufficient cooling water into the core in a LOCA situation.

When an “S” signal of safety injection is initiated, the safety injection pumps are started automatically to take water from the refuelling water storage pit located in the containment and inject coolant directly into the reactor vessel without passing through the loop.

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, together with the satety injection pump, cooling water until the core is reflooded. At the start of injection, cooling water is injected with a large flow rate and then, when the water in an accumulator has dropped to a certain level, the flow damper switches over the flow to a smaller flow rate which provides an injection flow similar to that given by the safety injection pumps on current plants.

Containment spray system

Four containment spray/residual heat removal pumps and four containment spray/residual heat removal coolers function as a containment spray system if a LOCA or main steam line break accident occurs.

When containment spray “P” signals are initiated, four containment spray/residual heat removal pumps are started automatically, and the stop valves in the pump discharge lines are opened automatically. The containment spray/residual heat removal pumps take water from the refuelling water storage pit and supply it to the containment spray header located at the top of the containment through the containment spray/ residual heat removal coolers.

This system has also a residual heat removal function used to remove decay heat from the core in normal cooling of plants and refuelling.

In-containment refuelling water storage pit

The refuelling water storage pit is formed in a horse shoe shape, and is located at the bottom level of the containment.

It provides a continuous source of water for the safety injection pumps and containment spray/residual heat removal pumps. Therefore, it is not necessary to switch over from the refuelling water storage pit to the containment recirculation sump as must be done on existing plants. During refuelling, the refuelling water storage pit is used also as a water source to fill the reactor cavity.

Emergency feedwater system

The auxiliary feedwater system, except for the emergency feed water storage pit, consists of two mechanical sub-systems. Each sub-system is provided with a motor driven auxiliary feedwater pump and a turbine driven auxiliary feedwater pump.

On receiving signals from the reactor protection system, the system starts feeding water automatically from the emergency feed water storage pit to the steam generator. If the steam generator heat transfer tubes, main feedwater pipes, or main steam pipes are broken, the system isolates the auxiliary feedwater to the damaged steam generator automatically by auxiliary feedwater isolation signals.

When the auxiliary feedwater system has been started and the plant has been stabilized at hot standby conditions after an accident or transient, the auxiliary feedwater system can be used to cool the plant to the temperature at which the residual heat removal system can be put in service. When that temperature is reached, the residual heat removal system is started to bring the plant to cold shutdown, and the auxiliary feedwater system is stopped.

Residual heat removal system

The residual heat removal system removes decay heat from the core by taking water from the hot legs of the primary cooling system by the four containment spray/residual heat removal pumps and returning the coolant to the cold legs of the primary cooling system through the four containment spray/residual heat removal coolers. The residual heat removal system has the capacity to cool the primary coolant temperature from 177 to 60°C within 20 hours after the reactor has been shut down.

4.3.6.3 Severe accidents (beyond design basis accidents)

Prevention of severe accidents

In the preliminary design of the APWR a safety system with four sub-systems has been adopted, the refuelling water storage pit (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 which bypasses the containment, the corresponding parts of the piping in the residual heat removal system are under consideration with a higher rating to prevent the interface LOCA from occurring since this type of accident can have very serious consequences to the environment.

Countermeasures during plant shutdowns

As safety enhancement during mid-loop operating mode, which are especially important among the countermeasures during plant shutdowns, some countermeasures are to be taken, such as adoption of RCS high water level operation, reinforcement of RCS water level indicators, automatic isolation of letdown line at low RCS water level, reinforcement of water injection functions during lowering of RCS water level, etc.

Also, as a precaution against the event of abnormal dilution of boric acid during the external power failure, interlocks to prevent boron dilution are under consideration.

Mitigation of severe accidents

In an APWR plant, as shown in Section 4.3.6.1, the mitigation of the consequences of a severe accident is also to be considered from the view point of risk reduction and greater protection. Specifically, as shown below, countermeasures against those events that threaten the integrity of the containment vessel are under consideration.

As the countermeasures against debris dispersion, the reinforcement of depressurization function of the primary system and the improvement of RV cavity form are considered countermeasures against damage by quasi-static over pressure, the normal containment vessel air recirculation system and an alternative containment vessel spray supplied from the fire service water system can be used. These systems can be used to cool the containment vessel and reduce the pressure if the containment vessel spray system is not available.

As countermeasure against containment vessel damage due to hydrogen combustion, a hydrogen control system (ignitors) can 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 RV cavity and water will be injected into the cavity from the fire service water system. Also a 1 m thick protective layer of concrete can be provided so that the containment vessel boundary is not exposed directly to the debris. Thus the molten debris will be coolable, and erosion of the concrete and overheating of the containment vessel atmosphere can be prevented. As countermeasures against the dispersion of debris, reinforcement of the primary system depressurization function and improvement of the reactor vessel cavity form are considered. It is also considered that the outlet from the RV cavity to the other containment vessel spaces should be constructed like a labyrinth.

4.3.7 Plant layout

The plant must be laid out so that the safety of the 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.

4.3.7.1 Buildings and structures, including plot plan

The standard arrangement is for a twin unit plant consisting of two reactor buildings, a turbine building, a common control building and auxiliary building.

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 which have relatively small 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 auxiliary 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.

The plant must be designed as follows for internal and external events such as jet aircraft, missiles, and fires.

Internal missiles: The design must be made in such a way that the safety of the reactor is not impaired due to the effects of internal missiles and broken pipes.

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 "Guidance for Verification of Fire Protection of LWR Facilities for Power Generation" in Japan.

(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, 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 major 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.

4.3.7.2 Reactor building

The reactor building consists of the reactor containment facility and the associated systems are installed. Figure 4.3-5 shows a cross-section of the reactor building.

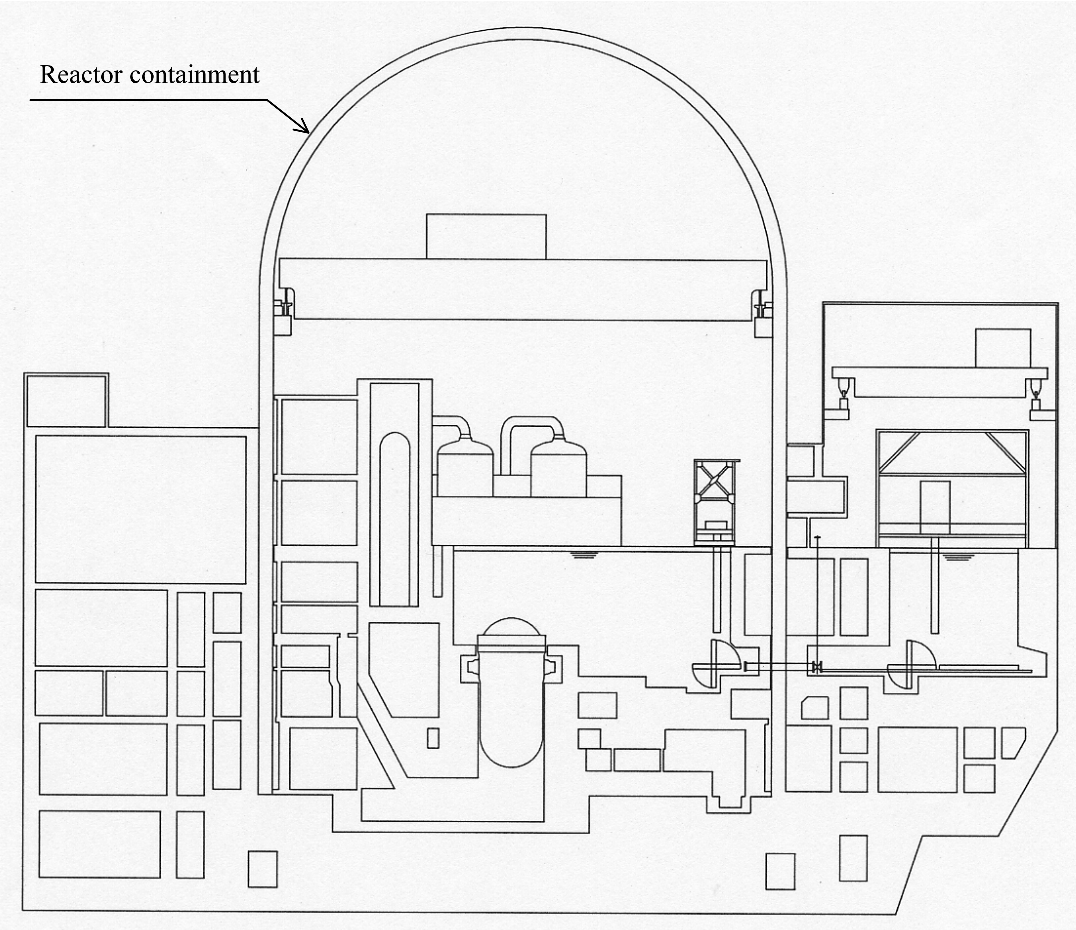

FIG. 4.3-5. APWR - Reactor building cross-section

4.3.7.3 Containment

The containment is part of the reactor containment facility and includes the internal concrete and the annulus compartment. The reactor containment facility is also part of the engineered safety systems which include the emergency core cooling system, the containment spray system, and the annulus air purification system, etc.

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 steel liner on the inner surface while the pressure withstanding function is provided by the concrete structure. An enclosed space (annulus compartment) surrounds 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 compartment.

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 $0.9 \times$ maximum design air pressure at normal temperatures.

The containment is provided so that the general public will not be affected by radiation if it leaks at this leak-rate even if the facilities related to the primary coolant system fail or are damaged. Therefore, severe accidents must also be carefully considred to ensure the integrity of the containment.

In current PWRs, 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 be distributed thinly 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.

4.3.7.4 Turbine building

The turbine generator, condensate and feedwater system auxiliary equipment, and other equipment are installed in the turbine buidling.

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 which 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.

4.3.7.5 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.

Control building

The control building which is common to both units, contains mainly the main control room, electrical equipment and access control equipment.

Auxiliary building

The Auxiliary building which is common to both units, mainly houses the radioactive waste treatment systems.

4.3.8 Technical data

General plant data

Power plant output, gross	1,538	MWe
Power plant output, net		MWe
Reactor thermal output	4,466	MWt
Power plant efficiency, net		%
Cooling water temperature		^{o}C

Nuclear steam supply system

Number of coolant loops	4	
Primary circuit volume, including pressuriser		m^3
Steam flow rate at nominal conditions	8.93×10^6	kg/h
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	-/-	^{o}C/MPa
Feedwater temperature/pressure		^{o}C/MPa

Reactor coolant system

Primary coolant flow rate [27.76 m^3/s]	77.3×10^6	kg/h
Reactor operating pressure	15.4	MPa
Coolant inlet temperature, at RPV inlet	approx. 289	^{o}C
Coolant outlet temperature, at RPV outlet	approx. 325	^{o}C
Mean temperature rise across core		^{o}C

Reactor core

Active core height	approx. 3.7	m
Equivalent core diameter	approx. 3.9	m
Heat transfer surface in the core		m^2
Fuel inventory	approx. 121	t U
Average linear heat rate	approx. 17.6	kW/m
Average fuel power density		kW/kg U
Average core power density (volumetric)	approx. 103	kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		

Fuel material	Sintered UO_2	
Fuel assembly total length	approx. 4,100	mm
Rod array	square, 17×17	
Number of fuel assemblies	257	
Number of fuel rods/assembly	264	
Number of control rod guide tubes		
Number of spacers		
Enrichment (range) of first core		Wt%
Enrichment of reload fuel at equilibrium core		Wt%
Operating cycle length (fuel cycle length)		months
Average discharge burnup of fuel		MWd/t
Cladding tube material	Zr base alloy	
Cladding tube wall thickness	approx. 0.6	mm
Outer diameter of fuel rods	approx. 9.5	mm
Overall weight of assembly		kg
Active length of fuel rods	approx. 3,700	mm
Burnable absorber, strategy/material		
Number of control rods assemblies (RRC)	69	
Number of grey control rods assemblies (GRC)		
Number of water displacer rods assemblies (WDR)		
Absorber rods per control assembly		
Absorber material: RCC	Ag-In-Cd	
Drive mechanism: RCC	Magnetic jack	
Positioning rate		steps/min [or mm/s]
Soluble neutron absorber		

Reactor pressure vessel

Cylindrical shell inner diameter	approx. 5 200	mm
Wall thickness of cylindrical shell		mm
Total height	approx. 13,600	mm
Base material: cylindrical shell		
RPV head		
Design pressure/temperature	/	MPa/^{o}C
Transport weight (lower part)		t
RPV head		t

Steam generators

Type	70F-1, U-tube heat exchanger	
Number	4	
Heat transfer surface	approx. 6,500	m^2
Number of heat exchanger tubes	5,380	
Tube dimensions	approx. 19	mm
Maximum outer diameter		mm
Total height	approx. 20,500	mm
Transport weight		t
Shell and tube sheet material		
Tube material	TT 690 alloy	

Reactor coolant pump

Type	100A, Single-stage, centrifugal pump	
Number	4	
Design pressure/temperature		MPa/°C
Design flow rate (at operating conditions) [6.94 m^3/s]	25,800	m^3/h
Pump head	approx. 91	m
Power demand at coupling, cold/hot		kW
Pump casing material	low alloy steel	
Pump speed		rpm

Pressuriser

Total volume		m^3
Steam volume: full power/zero power		m^3
Design pressure/temperature	/	MPa/°C
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter		mm
Total height		mm
Material		
Transport weight		t

Pressuriser relief tank

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Dry,	
Overall form (spherical/cyl.)	cylindrical, steel	
Dimensions (diameter/height)	approx. 45.5/69	m
Free volume		m^3
Design pressure/temperature (DBEs)	/	kPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate	<0.1	vol%/day
Is secondary containment provided?		

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage		kV
	rated capacity		MVA
Plant transformers,	rated voltage		kV
	rated capacity		MVA
Start-up transformer	rated voltage		kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)		6.6	kV
Number of low voltage busbar systems			
Standby diesel generating units: number			
	rated power		MW
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)	TC6F54	
Number of turbine sections per unit (e.g. HP/LP/LP)		
Turbine speed		rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/°C

Generator

Type		
Rated power	1,715	MVA
Active power		MW
Voltage	30	kV
Frequency		Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type		
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate		m^3 /s
Cooling water temperature		°C
Condenser pressure		hPa

Condensate pumps

Number	2	
Flow rate		kg/s
Pump head		
Temperature		°C
Pump speed		rpm

Condensate clean-up system

Full flow/part flow		
Filter type		

Feedwater tank

Volume		m^3
Pressure/temperature		MPa/ °C

Feedwater pumps

Number	3	
Flow rate		kg/s
Pump head		
Feedwater temperature		°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages	6	
Redundancies		

4.3.9 Measures to enhance economy and maintainability

In the APWR, top priority is given to the securement of safety and reliability, and at the same time, economic efficiency improvement is also made.

4.3.9.1 Reduction in construction cost

Through the effect of economies of scale achieved by making the plant output larger by about 30% compared with the existing 4-loop plant, the unit construction cost can be reduced.

In addition, the construction cost is also reduced by simplifying the ECCS system, such as dividing the safety system equipment into four sub-systems, adopting a high performance accumulator tank, installation of the refueling water storage pit inside the reactor containment vessel, as well as by reducing the amount of cables by adoption of optical multiplex transmission, adopting compact equipment, such as improved steam generators, plate heat exchangers, energy absorbing supports, making the building compact by rational arrangement of equipment through the utilization of 3D-CAD, etc.

4.3.9.2 Shortening of construction schedule

Shortening of the construction schedule is being studied by adopting super heavy-duty cranes, increasing the number of large piping modules and components, etc.

4.3.9.3 Shortening of maintenance outage

The regular mentainance outage can be shortened by adopting automatic and high-speed fuel handling system, bolting tensioning machine for reactor vessel opening and restoration, etc. Thus, the plant availability factor is expected to be improved.

4.3.9.4 Reduction in exposure to radiation

Reduction in employees' exposure to radiation can be expected by adopting the fuel assembly zircaloy grid, optimizing pH control in RCS by application of enriched boron, enhancement of purification capacity by increasing the flow rate for purification, etc. as the measures to reduce the source of radiation.

4.3.10 Project status and planned schedule

With the summarized technologies on PWR that have been improved based on the construction of PWRs and the experience in their operations, APWR has been improved remarkably on safety, reliability, operability, maintainability, and economic efficiency. It is expected to contribute largely to the supply of energy as the standard model of the PWRs which will be constructed in Japan including the Tsuruga-3 and 4.

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4.4 APWR+ (MITSUBISHI, JAPAN)

4.4.1 Introduction

The development of the advanced PWR (APWR) was started in the 1980s. The APWR is a summarization of PWR technologies in the 1980s and also includes advanced technologies from the present day. However, the conditions surrounding nuclear power have changed greatly since the initiation of development of the APWR, and the demand for improvements in economy is becoming particularly important.

For these reasons, Japanese PWR utilities (Kansai Electric Power, Hokkaido Electric Power, Kyushu Electric Power, Shikoku Electric Power and The Japan Atomic Power) and Mitsubishi Heavy Industries (MHI) have started the development of the next generation PWR (the APWR+).

The development target of APWR+ is aimed at more enhancements in economy, safety and reliability, reduction of the operators' work-load, and harmony with the environment based on conventional LWR and APWR technology. The target for starting APWR+ construction has been set for the end of the 2010s.

In the following sections, the features of this APWR+ are introduced.

4.4.2 Description of the nuclear systems

Table 4.4-I shows the design requirements for the APWR+ based on the above circumstances.

To enhance the economy, the plant capacity and the availability factor were increased from APWR. Enhancement of O&M and harmonization with the environment were adopted to meet the environmental conditions of the 21st century. Although the safety level of APWR is thought to be sufficiently high, further improvements for containment protection and mitigation during maintenance were considered.

TABLE 4.4-I. DESIGN TARGETS FOR THE APWR+

Items	Design Targets
Enhancement of economy	
Plant capacity	1700MWe class
Design availability factor	95%
Operation period	24 months max
Enhancement of Operation & Maintenance	
On Power Maintenance	applicable
Harmonization with environment	
MOX loading fraction	100% available
Enhancement of safety	
Core Damage Frequency	Lower than 10^{-7}/RY
CDF during maintenance	Lower than 10^{-7}/RY
Containment failure frequency	1/10 of CDF (Same level as APWR)

4.4.2.1 *Primary circuit and its main characteristics*

Table 4.4-II shows a comparison of some major parameters between APWR and APWR+. The plant capacity has been increased to the 1700 MW class and the economy has been enhanced by the advantage of scale. Enhanced plant capacity, long cycle operation, and reduction of the number of spent fuel assemblies are achieved by adopting long length fuel assemblies. The reactor vessel height has not been increased from APWR as a result of the simplification of lower core internals by adopting a top mounted in-core instrumentation system (ICIS). By adopting an advanced safety system in which expanded passive technology (decay heat removal using steam generator) is applied, simplification of the safety system design is accomplished. Increase in electric power output is possible by developing a large-sized reactor coolant pump and high-performance steam generator. By adopting 4-train configuration, on-power maintenance of safety equipments is available. This reduces the periodic inspection time and peak work-load during periodic inspection.

TABLE. 4.4-II. COMPARISON BETWEEN APWR AND APWR+

Items	APWR	APWR+
Electric power output (MWe)	1538	1750
Thermal power output (MWt)	4466	5000
RV outlet temperature (°C)	325	327
Primary system flow (m^3/h/loop)	25800	29100
SG steam pressure (MPa)	6.1	6.9
SG heat transfer area (m2)	6500	8350
Fuel assembly type	17x17	17x17
Number of fuel assemblies	257	257
Fuel effective length (m)	3.7	4.3
ICIS type	Bottom mounted	Top mounted
Neutron reflector	Stainless steel	Stainless steel
Height of Reactor vessel (m)	13.6	13.3
Inside diameter of Reactor Vessel (m)	5.2	5.2

4.4.2.2 *Reactor core and fuel design*

The reactor core consists of 257 17x17 fuel assemblies. A fuel assembly length of 4.3 m is used to meet the requirements for high output and long operating cycle. Also, the core is designed to use less than 5% enrichment fuel in 24-month operating cycles. The number of control rod clusters is reduced from 77 to 69 by adopting a high-performance control rod for 1/2 MOX cores. 100 % MOX cores are also available with 85 control rod clusters.

4.4.2.3 *Fuel handling and transfer systems*

The fuel handling and fuel transfer systems consist mainly of the refueling crane, fuel transfer system, and multi-functional mast type spent fuel pit crane as in existing plants. Considering the recent requirement to reduce the periodical inspection time, many improvements including an increase in the speed of each system have been introduced. Also, in order to reduce operations in radiation controlled areas, these systems can be operated automatically from a remote centralized control room instead of the present method of operating individually from a control station located next to each piece of equipment.

4.4.2.4 Primary components

(1) Reactor Vessel

Fig. 4.4-1 shows a comparison of the reactor vessel between APWR and APWR+.

The inside diameter of the reactor vessel is 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. The neutron irradiation of the vessel steel is reduced by a set of neutron reflectors, thus the reliability of the reactor vessel is increased.

(2) Reactor internals

As shown in Fig. 1, the reactor vessels of APWR and APWR+ have almost the same total height and outer diameter. The major difference between the internals of APWR and APWR+ comes from the length of the fuel assemblies. APWR loads 257 fuel assemblies with an effective length of 3.7 m, and APWR+ loads 257 assemblies with 4.3 m. The fuel allocation is the same. In order to contain longer fuel assemblies in the same sized reactor vessel with APWR, APWR+ adopts a simplified configuration with only one lower core support plate to support the weight of fuel. (In APWR, the weight of the fuel is transmitted and supported by a lower core plate and a lower core support plate.) This configuration enables to reduce of the total length of the core internals to be contained in the APWR reactor vessel. Such configuration also results in lower construction costs by producing more output using the same reactor vessel as APWR.

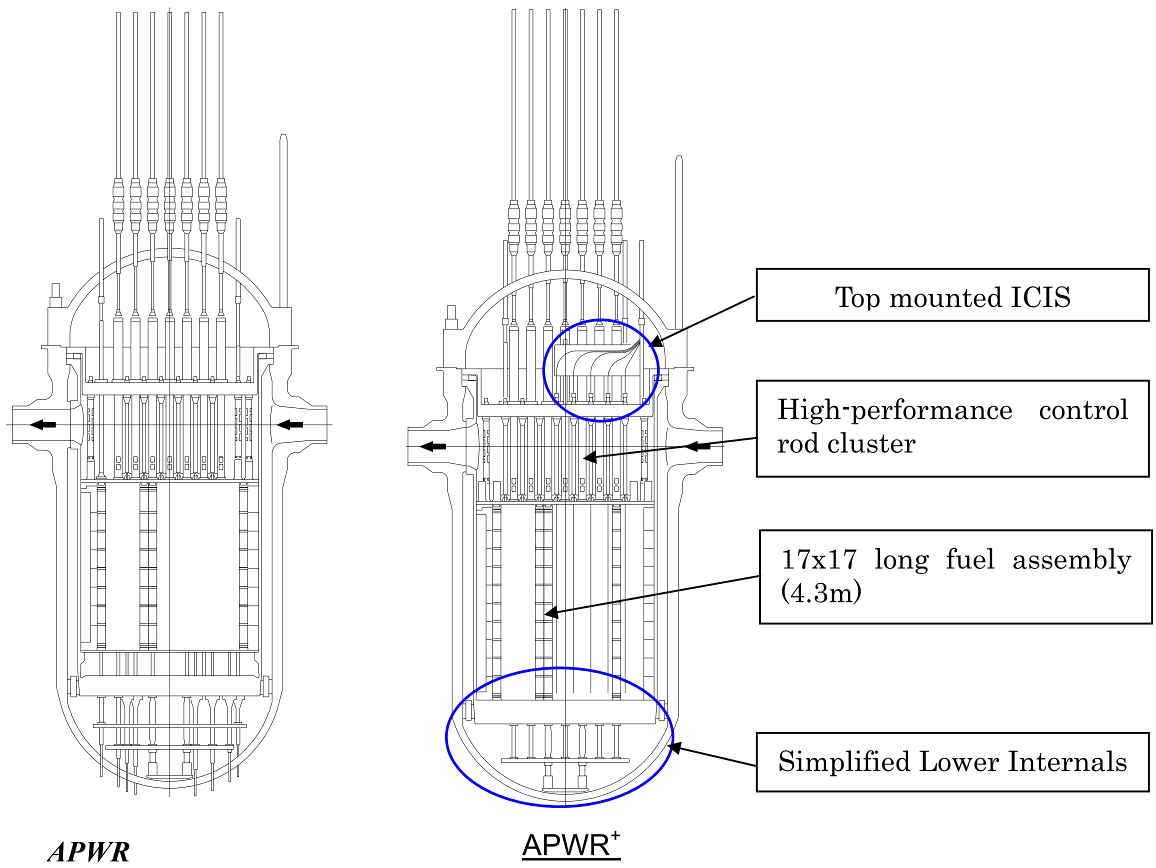


FIG. 4.4-1. Cross section diagrams of reactor vessel

(3) In-core Instrumentation System (ICIS)

The top mounted ICIS, which is a mechanism for introducing the detectors into the core from the top of the reactor vessel, is selected for APWR+. Fixed in-core detectors are selected as the in-core neutron detectors.

Several detectors are fixed on a plate and handled together as an assembly unit. This method aims to shorten the handling time and to reduce the radiation exposure of workers during refueling compared to the method where the detectors are handled one by one. Such assembled configuration of detectors is called “top mounted instrumentation assembly”. Fig. 4.4-2 shows a top mounted instrumentation assembly. These assemblies are installed in the top plenum of the reactor vessel.

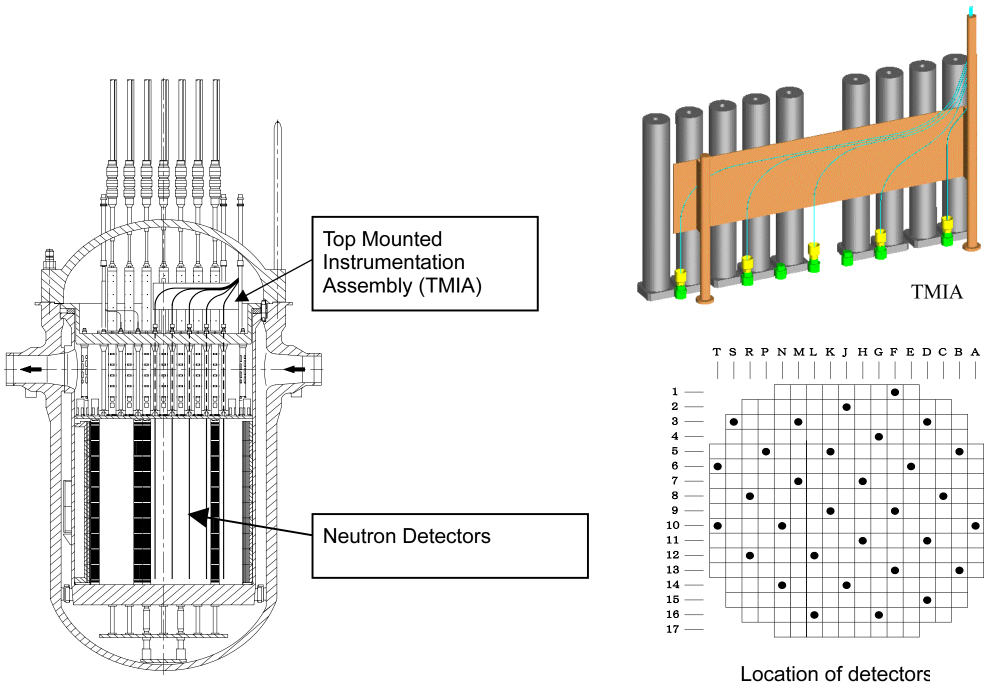


FIG. 4.4-2. Top mounted in-core instrumentation system

In this top mounted instrumentation assembly configuration, all detectors in the assembly unit are assembled and penetrate the reactor vessel head through one penetration. Therefore, detectors are not straight and need to bend to reach the core through the top plenum. This route is conducted on the top mounted instrumentation assemblies in the top plenum. The design of the support columns penetrating the reactor vessel head is the same design of the thermocouple conduit support columns of conventional LWRs. According to this, the maximum number of detectors for one top-mounted instrumentation assembly is 6 due to the limitation of the diameter of the support column.

(4) Steam Generator

The steam generator of APWR+ has increased capacity based on the large type steam generators (Type 70F-1) of the APWR.

In order to prevent increase of the containment size, the increase in size of the steam generator has been minimized. The outside diameter of the lower shell is almost the same as the APWR steam generator (Type 70F-1) because the thin tube and triangular arrangement have been adopted for the heat-transfer tubes. The height of the steam generator is achieved because the heat-transfer area is minimized by adopting the economizer.

Table 4.4-III shows a comparison of some major parameters between APWR and APWR+.

TABLE 4.4-III. COMPARISON OF MAIN SPECIFICATION AT STEAM GENERATOR

	APWR (70F-1 Type)	**APWR+**
Thermal Output (MWt/SG)	Approx.1120	Approx.1250
Steam pressure (MPa)	6.1	6.9
Reactor coolant inlet temperature (°C)	325	327
Heat transfer area (m^2)	6500	8350
Economizer	-	Axial economizer
Tube arrangement	square	triangular
Height (m)	20.6	20.9
Upper shell diameter (m)	4.8	5.1
Lower shell diameter,(m)	3.9	3.9

(5) Reactor Coolant Pump (RCP)

The primary coolant pumps are the vertical type, single stage, and slanting style pumps. The drive motors are attached to the pump 3-phase air-cooling induction motors.

- Design flow rate: 29100 m^3/h/pump
- Design pump head: 126.5 m

4.4.2.5 Reactor auxiliary systems

The chemical and volume control system has the following main functions.

The first function is to adjust the amount of water contained in the reactor coolant system. In normal operation, the letdown and charging flows are controlled so that the water level in the pressurizer is maintained at the programmed level. At the same time, seal water is injected into the primary coolant pump seals.

The second function is to control the boron concentration and the water chemistry of the primary coolant system. The concentration of boron in the primary coolant system is adjusted by adding make-up water from the reactor make-up system or boric acid solution from the boric acid water tank as required to compensate the reactivity change with fuel burn-up or load changes.

The third function is to purify the primary coolant. The primary coolant is purified by a demineralizer and filter in the let down line. The let down flow is taken from the cross-over leg of the primary coolant system, and the coolant is cooled by the regenerative heat exchanger and let down heat exchanger, and then purified in the demineralizer. The water chemistry of the primary system can be controlled by adding hydrazine or lithium hydroxide, passing the water through a cation demineralizer, and adding hydrogen gas to the vapor space of the volume control tank.

To supply water to the primary coolant system and seal water to the primary coolant pumps, two charging pumps are used taking water from the volume control tank.

4.4.2.6 Operating characteristics

The reactor is designed to be operated automatically within the range of 15 to 100% of rated output by the reactor control system. Even in the low output range below 15%, the control rod control system can control the reactor automatically in the low power operating mode. The primary coolant average temperature is controlled to a programmed value that increases linearly with the turbine output.

4.4.3 Description of the turbine generator plant systems

4.4.3.1 Turbine generator plant

The turbine consists of a high-pressure turbine and three double flow type low-pressure turbines with reaction blading. The last stage blades are 54 in. long. The moisture separator/reheater has a two-stage heater to achieve high efficiency.

4.4.3.2 Condensate and feedwater systems

The configuration of feed heaters in the condensate and feedwater systems are six stages which consist of four stages low pressure heaters, the deaerator and single stage high pressure heaters. The pumps installed in the condensate and feedwater systems consist of the main feedwater pumps and condensate pumps with booster pumps. Even if one pump fails, normal operation can be maintained.

4.4.3.3 Auxiliary systems

No information provided.

4.4.4 Instrumentation and control systems

4.4.4.1 Design concept, including control room

The main control room is provided with compact consoles on which CRTs and flat display panels are mounted.

The plant is operated by touch screen operations, and the monitoring information necessary for operation is displayed on the same screens that are used for the plant operation. 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.

4.4.4 2 Reactor protection and other safety systems

The reactor protection system and other safety systems are functionally distributed digital systems.

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 four trains. Each train has digital devices. To interface these systems with the auxiliary equipment in the plant, remote input/output devices arranged collectively in the plant are connected to the host computer through optical fiber cables.

The reactor protection system and other safety systems are provided with automatic test equipment so that periodical tests can be conducted fully automatically.

4.4.5 Electrical systems

4.4.5.1 Operational power supply systems

The operational AC power supply system can receive external power from the main power supply system and from the stand by 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. Therefore, the operational AC power supply system can receive power through the plant transformers continuously 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 at hot standby conditions and enable it to be shut down safely.

The buses of the operational AC power supply system are divided into two main groups: the 6.6 kV high voltage system and the 440 V low voltage system, each comprising normal buses to supply power to loads such as primary coolant pumps, feed pumps and other equipment required for normal plant operation; and four-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 these 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, and an instrumentation and control power supply (consisting mainly of inverters) for supplying power to the instrumentation and control equipment which are mainly computer loads.

4.4.5.2 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 generators and battery equipment. The emergency generators will start automatically immediately after an accident occurs or external power is lost, and supply power to the emergency buses. As shown Fig. 4.4-3, gas turbine generators are adopted in addition to conventional diesel generators for emergency power. The configuration of emergency power has redundancy and diversity.

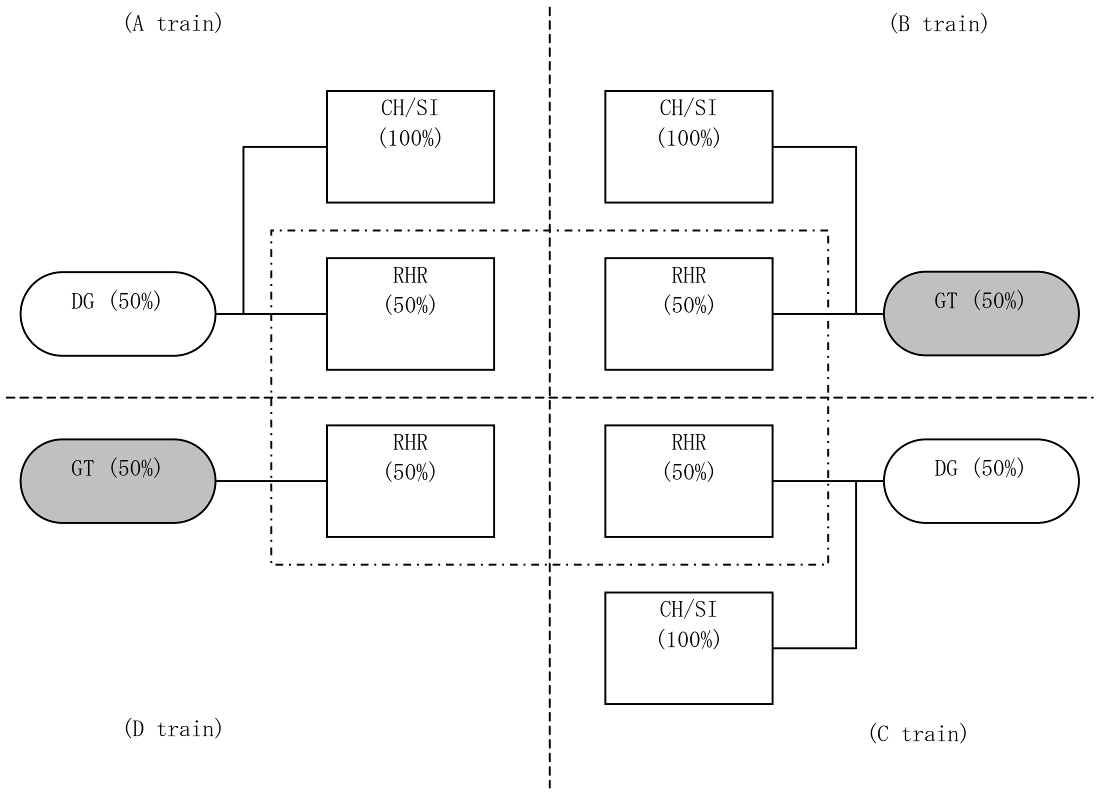


FIG. 4.4-3. Four-train and emergency generators

The DC power supply system can supply power to the instrumentation control power system following an instantaneous power failure, and also has sufficient capacity to supply the switchgear, which must operate following a loss of external power, and to supply the excitation current for the diesel generators. Also, 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 four trains to be consistent with the four trains and four sub-systems, and the instrumentation and control power system is divided into four trains to be consistent with the four channels.

4.4.6 Safety concept

4.4.6.1 Safety requirements and design philosophy

As a new concepts, APWR+ considers the following:

1) Adoption of a passive safety system

By adopting the following concepts, safety, reliability, and economy are enhanced.

- Core cooling using steam generator during accident
- Adoption of advanced boric injection tank

2) Train configuration

As on-power maintenance is possible by adopting 4-train - 4 subsystems, safety, economy, and operation & maintenance are enhanced.

3) Introduction of equipment with the diversity design

By diversifying the emergency power supply, safety, reliability and economy are enhanced.

Fig. 4.4-4 shows the features of APWR+ safety related systems.

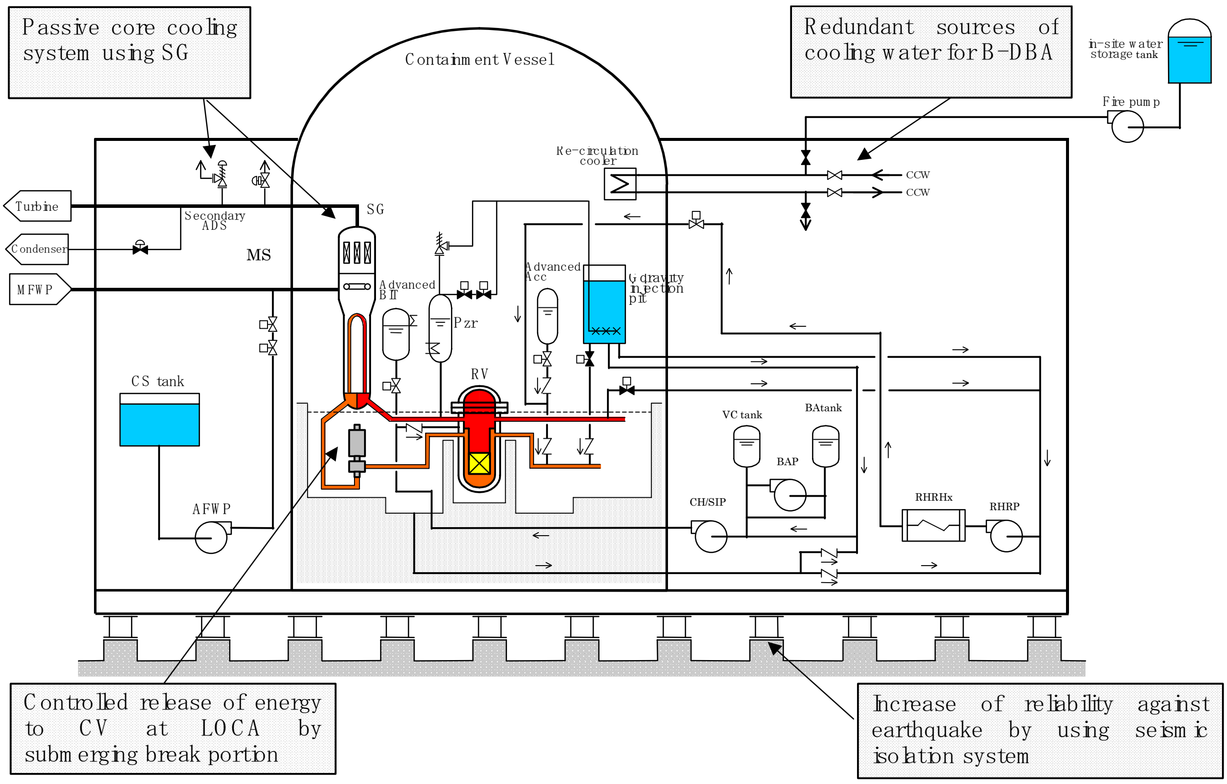
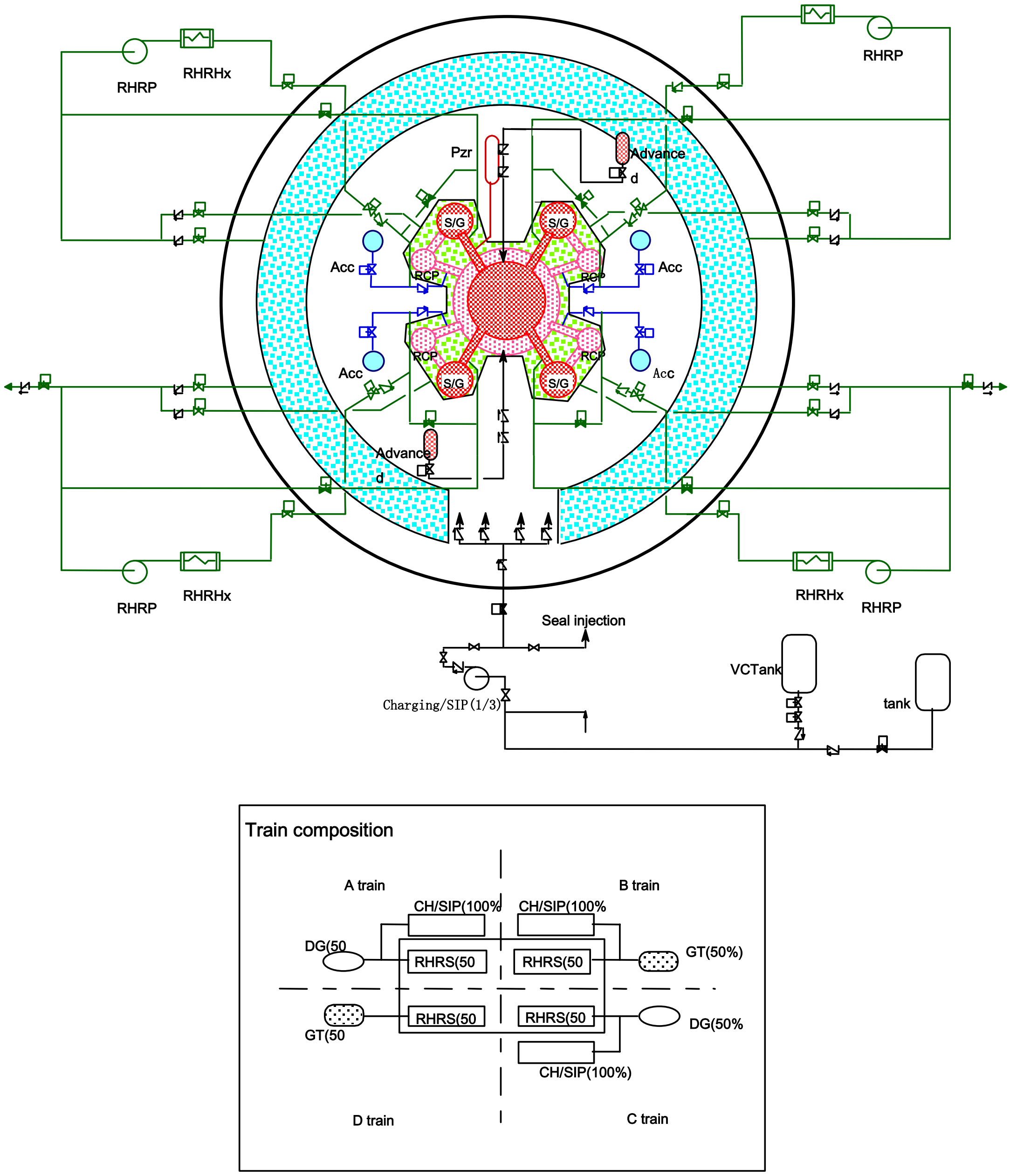


FIG. 4.4-4. Features of APWR+ safety related systems

4.4.6.2. *Safety systems and features (active, passive and inherent)*

The design requirements relating to safety systems are reduction of the construction cost, on-power maintenance and enhancement of safety. To meet these requirements, the safety system configuration shown in Fig. 4.4-5 was adopted. The features of APWR+ safety systems are as follows:



DG:Diesel generator GT:Gas turbine generator
CH/SIP:Charging/Safety injection pump
RHRS:Residual heat removal system

FIG. 4.4-5. Safety system configuration

1) Safety system configuration

To meet the design requirement for on-power-maintenance, the system configuration has been changed from 2 train - 4subsystems to 4 train - 4 subsystems. By introducing this train configuration, on-power maintenance can be conducted during the whole time of operation. This will result in the reduction of the time duration for periodic maintenance by about 3 days. Although the number of electrical systems has increased, the power production cost will be decreased because of enhancement of the plant availability. In addition, reliability during maintenance will be improved, because the number of stand-by safety equipment is increased.

2) Equipment diversity

To meet the design requirements for core damage frequency, diversities of safety systems are introduced. For emergency power supply systems, 2 gas turbine generators are introduced in addition to 2 diesel generators. By introducing the gas turbine generators, the reliability of emergency power supply systems is improved because of the reduction of the probability of common mode failure. In addition, since the support systems are not required for a gas turbine, the system configuration is simplified.

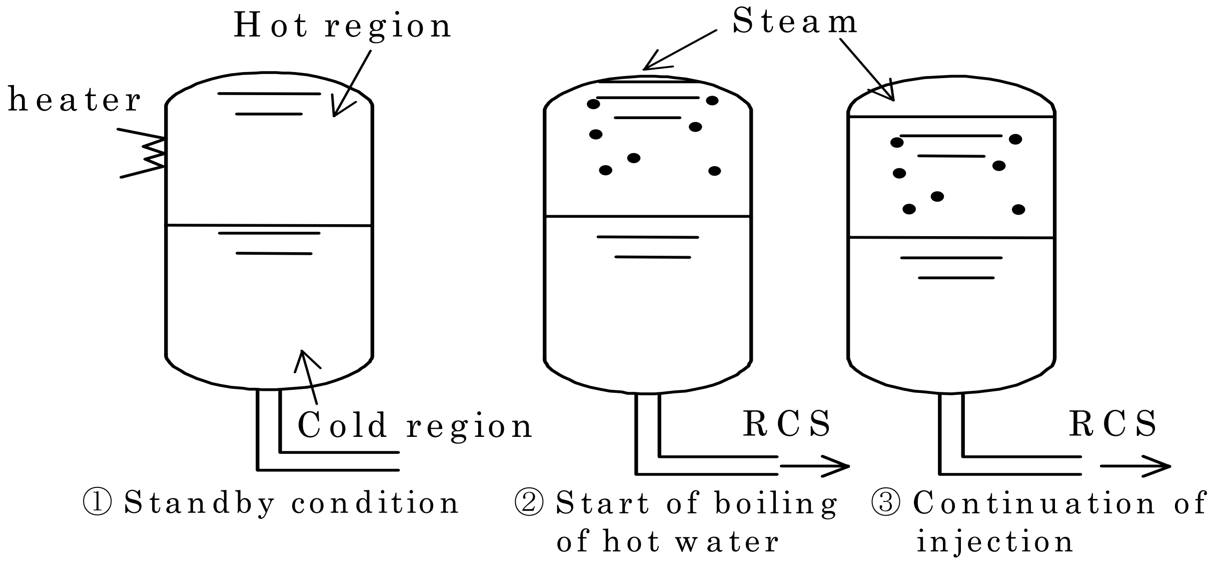
3) Core cooling system using steam generator

A new emergency core cooling concept, in which steam generator cooling is applied, is adopted in APWR+. To remove the decay heat rapidly and reduce the primary system pressure during small LOCA, a secondary depressurization system is introduced. By this system, high head safety injection pumps are eliminated, since the primary system pressure is rapidly reduced and the core will not be uncovered during small break LOCA by supplying water from the advanced accumulators and early start-up of the low head safety injection pumps. Since reactor vessel is filled with water by supplying water with low head safety injection pumps, long-term decay heat removal is achieved.

Since a large amount of water is injected into the primary system during large LOCA by the break of a cold leg or a cross over leg pipe, the break point of the reactor coolant system (RCS) will be flooded within a short period. After that, decay heat will not be released into the atmosphere in the containment vessel. Injection water by low head safety injection pumps will reach the break point certainly through the core during large LOCA by the break of a hot leg pipe or a pipe around the pressurizer. With the reduction of decay heat, boiling at the core is stopped and overflow of water from the break point changes water single-phase flow from two-phase flow. Afetr that, the decay heat will not be released into the atmosphere in the containment vessel. From the above, containment spray systems is eliminated. These new concepts contribute the simplification of the safety systems.

4) Advanced Boron Injection Tank (ABIT)

An advanced boron injection tank (ABIT), through which the boric acid is passively injected to the core was developed in order to control the reactivity in the core during an accident. In the upper part of the ABIT, heated boric acid solution is stored. When the system pressure decreases to below the saturation temperature of this heated boric acid solution, flashing of the water will occur and then the boric acid solution in the lower part of the tank will be injected into the reactor coolant system. Fig. 4.4-6 shows the concept of the advanced boron injection tank. The validity of this newly designed tank has been confirmed by model tests. By introducing ABIT, the reliability of reactivity control during cool-down transients, such as steam line break accidents, is improved.



ABIT CONFIGURATION

	Hot region	Cold region
Pressure	Approx. 17MPa	
Temperature	325 °C	< 280 °C
Volume	12.7m^3	12.7m^3

FIG. 4.4-6: Advanced boron injection tank

4.4.6.3 Severe accidents (beyond design basis accidents)

Advanced safety systems with the diversified design and RCS flooding design are adopted for beyond design basis accidents.

Water to the containment re-circulation coolers is supplied from Component Cooling Water System (CCWS) in the case of APWR. In the APWR+ design, water can be supplied from both CCWS and an on-site water storage tank. This will reduce the conditional containment failure frequency of APWR+.

The core is cooled by supplying water by the low head safety injection pumps at design basis accidents. If the operation failure of the low head safety injection pump is occurred, water in the refueling water storage pit (RWSP) installed on the operating floor will fill into the core by gravity with operating valves. Therefore, it is possible to obtain a margin of time for operators. If water in RWSP could not be injected to the core, core melt will be occurred. In this case, since water in RWSP can also flood RCS, in-vessel retention can be achieved.

4.4.7 Plant layout

The plant must be laid out so that the safety of the reactor facilities is not impaired, and the exposure dose around the plant is below a specified limit. In addition, the separation of redundant trains, earthquake resistance, and maintenance of the safety system equipment must be considered to give an optimum arrangement.

4.4.7.1 Buildings and structures, including plot plan

The arrangement is for a twin unit plant consisting of a reactor building and a turbine building. The reactor building is an integrated primary building and consists of two reactor buildings, a common control building and a common waste building, which are found in conventional PWR plants. The buildings have been integrated for adoption of the isolated system. The isolated system makes standard design of the reactor building and reduction of the construction resources.

The 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 into the environment if an accident occurs, and those that have serious consequences for the plant.

Class B: Those which have relatively small 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.

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.

The plant must be designed as follows for internal events such as jets, missiles and fires.

Jet 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.

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 designed so 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 in such a way 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.

4.4.7.2 Reactor building

The reactor containment facility, the fuel handling systems, the instrumentation systems, waste disposal systems and their associated systems are installed in the reactor building.

4.4.7.3 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 include the emergency core cooling system and the annulus air purification system.

The containment system is designed to suppress or prevent the possible dispersion of 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 feed-water system.

The leakage prevention function of the containment is provided 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 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 ´ maximum design air pressure at normal temperatures.

The containment is provided so that the general public will not be affected by radiation if it leaks at this leak-rate 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 an APWR, the refueling water, which is the water supply used after an accident, was stored in a pit at the bottom of the containment. In the APWR+, the refueling water is stored in a pit on the operation floor to enable charging of the cooling water by gravity.

4.4.7.4 Turbine building

The turbine building is a common building for two unit plants and the reduction of its building volume is aimed at by sharing an equipment access area and so forth.

4.4.7.5 Other buildings

No information provided.

4.4.8 Technical data

General plant data

Power plant output, gross	1,750	MWe
Power plant output, net		MWe
Reactor thermal output	5,000	MWt
Power plant efficiency, net	35	%
Cooling water temperature		°C

Nuclear steam supply system

Number of coolant loop	4	
Primary circuit volume, including pressuriser	Approx. 500	m^3
Steam flow rate at nominal conditions	Approx. 3,000	kg/s
Feedwater flow rate at nominal conditions	Approx. 3,000	kg/s
Steam temperature/pressure	298 / 7.2	°C/MPa
Feedwater temperature/pressure	235 / 7.2	°C/MPa

Reactor coolant system

Primary coolant flow rate	24.1×10^3	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	291	°C
Coolant outlet temperature, at RPV outlet	327	°C
Mean temperature rise across core	36	°C

Reactor core

Active core height	4.3	m
Equivalent core diameter	3.9	m
Heat transfer surface in the core	Approx. 8,700	m^2
Fuel inventory	141	tU
Average linear heat rate	16.7	kW/m
Average fuel power density	36	kW/kgU
Average core power density (volumetric)	98	kW/l
Thermal heat flux, F_q	39	kW/m^2
Enthalpy rise, F_H	207	kJ/kg
Fuel material	UO_2	
Fuel assembly total length	4900	mm
Rod array	square, 17×17	
Number of fuel assemblies	257	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	24	
Number of spacers	11	
Enrichment (range) of first core	<5	wt%
Enrichment of reload fuel at equilibrium core	4.95	wt%
Operating cycle length (fuel cycle length)	18-24	months
Average discharge burn-up of fuel	Approx. 6,000	MWd/t
Cladding tube material	Zr base Alloy	
Cladding tube wall thickness	0.6	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly		kg
Active length of fuel rods	Approx. 4,300	mm
Burnable absorber, strategy/material	Integrated / Gd	
Number of control rods assemblies (RCC)	69	
Number of grey control rods assemblies (GRC)	0	
Number of water displacer rods assemblies (WDR)	0	
Absorber rods per control assembly	24	
Absorber material:	Ag-In-Cd + enriched B_4C	
Drive mechanism:	Magnetic jack	
Positioning rate	72	steps/min
Soluble neutron absorber	Boric Acid (enriched ^{10}B)	

Reactor pressure vessel

Cylindrical shell inner diameter		5,200	mm
Wall thickness of cylindrical shell		260	mm
Total height		13,300	mm
Base material:	cylindrical shell	low-alloy steel	
	RPV head	low-alloy steel	
Design pressure/temperature		17 / 343	MPa/°C
Transport weight (lower part)			t
	RPV head		t

Steam generators

Type	U-tube heat exchanger	
Number	4	
Heat transfer surface	8350	m^2
Number of heat exchanger tubes	7880	
Tube dimensions	17.5	mm
Maximum outer diameter	53,400	mm
Total height	20,900	mm
Transport weight		t
Shell and tube sheet material	SFVQ1B	
Tube material	TT 690 alloy	

Reactor coolant pump

Type	Single-stage, centrifugal pump	
Number	4	
Design pressure/temperature	17/343	MPa/°C
Design flow rate (at operating conditions)		kg/s
Pump head	126.5	m
Power demand at coupling, cold/hot		kW
Pump casing material		
Pump speed		rpm

Pressuriser

Total volume	73	m^3
Steam volume: full power/zero power	27/54	m^3
Design pressure/temperature	17/360	MPa/°C
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter		mm
Total height		mm
Material		
Transport weight		t

Pressuriser relief tank Not applicable

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Dry,	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	49/67.5	m
Free volume	95,000	m^3
Design pressure/temperature (DBEs)	490/	kPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate	<0.1	vol%/day
Is secondary containment provided?		

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage		kV
	rated capacity		MVA
Plant transformers,	rated voltage		kV
	rated capacity		MVA
Start-up transformer	rated voltage		kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)		6.6	kV
Number of low voltage busbar systems			
Standby diesel generating units: number		2	
	rated power	5	MW
Standby gas turbine generating units: number		2	
	rated power	5	MW
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)	TC6F54	
Number of turbine sections epr unit (e.g. HP/LP/LP)		HP/LP/LP/LP
Turbine speed		rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/°C

Generator

Type	
Rated power	MVA
Active power	MW
Voltage	kV
Frequency	Hz
Total generator mass	t
Overall length of generator	m

Condenser

Type	
Number of tubes	
Heat transfer area	
Cooling water flow rate	m^2
Cooling water temperature	m^3/s
Condenser pressure	hPa

Condensate pumps

Number	
Flow rate	kg/s
Pump head	kg/s
Temperature	°C
Pump speed	rpm

Condensate clean-up system

Full flow/part flow	
Filter type	

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/°C

Feedwater pump

Number	
Flow rate	kg/s
Pump head	
Feedwater temperature	°C
Pump speed	rpm

Condensate and feedwater heaters

Number of heating stages	
Redundancies	

4.4.9 Measures to enhance economy and maintainability

The followings show a summary of measures to enhance the economics and maintainability.

To simplify the design

The followings are the APWR+ features to simplify the design, compared with conventional PWRs.

- Top-mounted ICIS eliminates nozzles at the bottom of the reactor vessel.

- The high head safety injection pumps are eliminated, since primary pressure is depressurized by using steam generators in the early stage of small LOCA.

- The containment spray system is eliminated. During large LOCA, the break point of the reactor coolant system (RCS) will be flooded within a short period and the decay heat will not be released into the atmosphere in the containment vessel.

To improve the operation flexibility

The followings are the APWR+ features to improve the operation flexibility.

- Low power density core utilize fuel effectively under 24 month operation.
- The core design of 100% MOX loading is applicable.
- On-power maintenance is achieved by adopting 4 trains - 4 subsystems.

To reduce the cost of equipment and structures

The followings are the APWR+ features to reduce the cost of equipment and structures.

- The reactor vessel size of APWR+ is the same as APWR by adopting simplification of lower reactor internal structure, long fuel (4.3 m) and top-mounted ICIS, although the output power is increased from APWR.
- Adoption of gas turbine generators eliminates auxiliary equipments for generators, compared with diesel generators since gas turbine generators could be cooled by air.
- Seismic isolation system reduces volume of concrete, supports for pipe, duct, cable tray, and so forth.

To reduce the construction period

No information is provided.

To reduce the scope of maintenance during operation and outage

On-power maintenance reduces outage time and peak work-load during outage.

To make the maintenance easier and with lower radiation exposure

No information is provided.

To increase the plant availability and load factor

95 % plant availability is achieved by 24 month operating cycles and less than 27 day periodic inspection.

To reduce the power generation cost

The followings are the APWR+ features to reduce the power generation cost.

- The output power of APWR+ (1,750MWe) is increased 14% compared with APWR;
- The system configuration and equipments are simplified as described above;
- 95 % plant availability is achieved as described above.

4.4.10 Project status and planned schedule

No information is provided.

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4.5 BWR 90+ (WESTINGHOUSE ATOM, SWEDEN)

4.5.1 Introduction

Westinghouse has continued its BWR development with an "evolutionary" design of nominally 1500 MWe, called the BWR 90+. The new design represents a further development of the BWR 90. That design built closely on the advanced BWR 75 design, which formed the basis for the construction of six well performing nuclear power plant units in Finland and Sweden.

The aim of the BWR 90+ program was to maintain and develop a competitive BWR option for the anticipated revival of the market for new power plants in the 21st century. The design should offer a reliable power generation at reduced construction and operation costs and incorporates significant safety improvements. An easily understandable level of enhanced safety was important in order to attain public acceptance.

With respect to economy, the work was aimed at developing a plant with reduced investment cost, short construction time, and high-energy availability. The fundamental design features of previous designs with respect to power production were incorporated.

The most significant improvements result from a modified containment design. It provides an inherent protection of the cooling of the reactor core against possible loss-of-coolant-accidents during a refuelling outage by flooding. The effects of severe accidents have been taken into account, and the containment structure will not be the first barrier to arrest a molten core. The improved design, and utilisation of modular construction procedures, will render a significant reduction in costs and construction time.

A "suitable" plant design involves many different aspects - the design of various systems, choice of materials and components, their installation, radiation shielding, accessibility to components, transport routes, proper routing of ventilation air, general building arrangement, etc. The end result will always represent a compromise between a number of aspects. In this context, a co-operation with the Finnish and Swedish operators Teollisuuden Voima Oy (TVO), Forsmark Kraft AB and OKG with their feedback of practical experience, has been of great value for the development of BWR 90+.

Some of the special features of the BWR 90+ are briefly reviewed below.

4.5.2 Description of the nuclear systems

The design of BWR 90+ was guided by careful considerations regarding flexibility and reliability, in particular with respect to the energy production capability. In this context, the following two design guidelines have been of paramount importance:

- Reliable electricity production shall be ensured by adopting proven system design and components;
- Moderate development steps are introduced only when bringing improvements.

The basic design of most of the reactor pressure vessel internal parts is maintained from the BWR 75 design, i.e., they are not welded to the reactor pressure vessel, and may easily be removed at refuelling, yielding time savings. Apart from steam separators of an improved design, and that the core spray nozzles have been removed; the arrangement of the internals is quite similar to that of the predecessors.

The recirculation system is based on the proven design of internal glandless pumps driven by wet asynchronous motors, supplied with variable frequency - variable voltage power. This type of pump has been operating reliably for more than four million operating hours since 1978 in Westinghouse BWR plants. The internal pumps eliminate the risk of pipe ruptures below the top of the core, and provide means for rapid and accurate power control; they are advantageous for load following purposes.

Another proven design feature is the Westinghouse 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 a capability of rapid recirculation flow-rate reduction, it provides an efficient ATWS (Anticipated Transient Without Scram) countermeasure.

Compared with its predecessor, the BWR 90+ design is characterised by an increased reactor core size. In addition, there is a distinct change to increased diversity and use of passive features to achieve safety functions, and an improved separation between safety and non-safety functions and systems. An auxiliary condenser is provided as a backup to the ultimate heat sink, and a boron injection system driven by stored pressure replaces the traditional system with piston pumps. Furthermore, the BWR 90+ has a primary containment with uncovery modified configuration. The new containment design offers protection against core uncovery in the event of a Loss-of-Coolant-Accident (LOCA) during refuelling operations, and significantly improved mitigation of the potential consequences of a severe accident with a release of core melt material from the reactor pressure vessel. Besides, it facilitates construction activities and contributes to reduced construction time and costs.

As noted above, major portions of the energy-production part of the BWR 90+ plant will be quite similar to its predecessor, and for that reason most of the development efforts have been focused on safety systems and their performance.

4.5.2.1 Primary circuit and its main characteristics

The general reactor pressure vessel arrangement is very much the same as in the Forsmark 3 and Oskarshamn 3 plants in Sweden.

The recirculation system is based on the use of internal glandless pumps which provides means for an accurate control of the reactor power, and eliminates large break LOCAs below the core level.

The "dried" steam is conveyed from the RPV to the turbine plant through four steam lines. The steam lines connect to nozzles with built in "flow limiters", evenly distributed along the vessel circumference; own medium operated isolation valves are provided on the inside and outside of the containment wall, the outer valve is equipped also with a motor operated actuator to ensure tightness after closure.

The feedwater enters the containment via two lines, each with inner and outer isolation valves, splitting up into four lines adjacent to the RPV for connection to four nozzles, at "mid-height" of the vessel. The nozzles and the internal removable feedwater distributors are of a special Westinghouse design that ensures a "thermal sleeve" protection against the "cold" feedwater for the RPV wall, and an efficient distribution into the downcomer.

The RPV is provided with a pressure relief system, which consists of 16 safety (relief) valves connected evenly onto the four steam lines, with blowdown pipes leading down into the condensation pool.

4.5.2.2 Reactor core and fuel design

4.5.2.2.1 General

The reference core for the BWR 90+ is a 4250 MWth equilibrium core with fuel assemblies of the SVEA-96 Optima2 type. The reactor core is largely based on the previous Westinghouse BWR design and advancements in fuel design. In comparison with the previous reactor generation, the main difference lies in the radial extension of the core. The power generation per fuel assembly has been reduced.

The core consists of 872 fuel assemblies. Each fuel bundle is located in a fitted fuel channel that serves to guide the coolant flow along the fuel rods as well as to give mechanical support and desired geometric confinement in the core. 213 cruciform control rods are manæuvered in water gaps between the fuel channels.

4.5.2.2.2 Core characteristics

The active core height is 3710 mm, and the equivalent core diameter is 5156 mm.

The core flow is in the range of 14500 - 17250 kg/s at full power operation. A stable flow condition is established by throttling the coolant inlet at core support by an orifice for most of the fuel assemblies.

At the rated power of 4250 MW_{th}, the average fuel assembly power is 4,874 MW, and the power density is 54,9 kW/l, corresponding to a specific power of 27.8 kW/kgU.

Flow control is possible from 100 % down to about 60 % of rated power (cf. Figure 4.5-1).

4.5.2.2.3 SVEA-96 Optima2

The SVEA-96 Optima2 assembly consists as other SVEA fuel designs of four sub-bundles, one handle part and a channel. The sub-bundles are separated by a double-walled cross in the channel, which in SVEA-96 Optima2 forms five flow channels, one square center channel and four rectangular gaps in the cross wings.

Each sub-bundle consists of 24 fuel rods in a 5x5-1 lattice. Three of the fuel rods in each sub-bundle are part length rods and two of them (2/3 in length) are placed adjacent the central channel and the third (1/3 in length) is placed in the outer corner. All rods have the outer diameter increased to 9,84 mm.

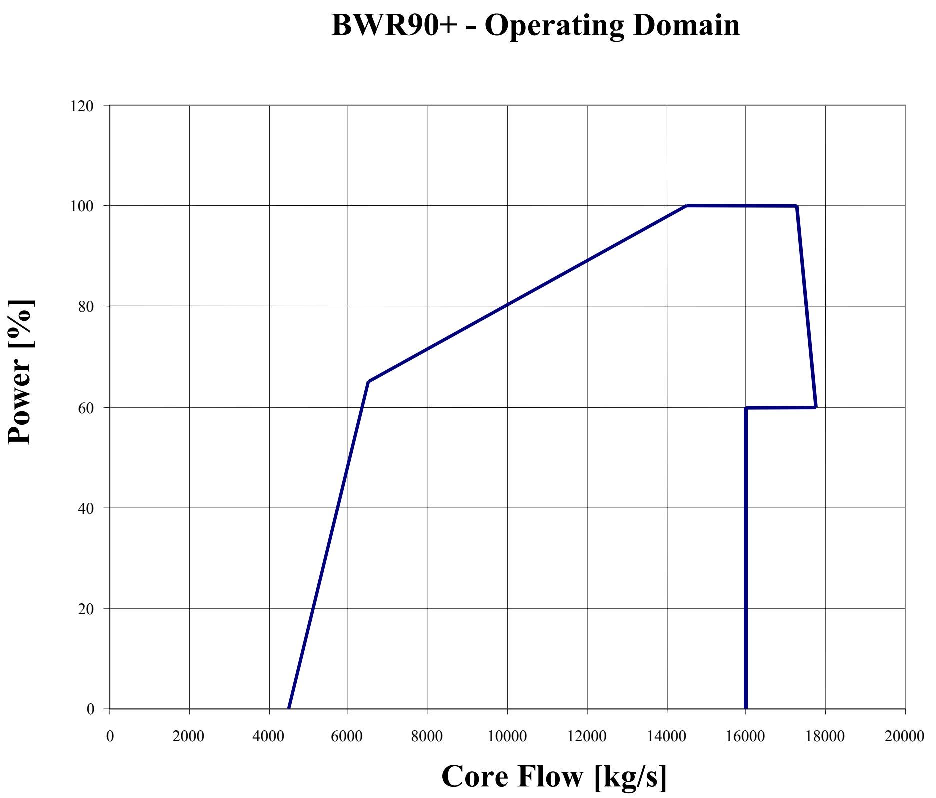

FIG. 4.5-1. Flow and power control range

The reduction of UO_2 in the upper portion of the fuel improves shut down margin considerably. Hence, there will not be any foreseeable need for axially distributed enrichment (the natural uranium end nodes disregarded). In addition, the demand of any advanced utilisation of a burnable absorber (BA) will consequently be alleviated.

A more roomy upper part of the fuel is beneficial with regard to pressure drop and hydraulic stability, which is favoured not only by the pressure drop gain per se but also by the net downwards shift in axial profile of pressure drop, including the pressure drop over the eighth spacer.

4.5.2.2.4 Stability

Sufficient margin against hydrodynamic instability in the coolant flow through the individual fuel channels is mainly ensured by having a low ratio of two-phase to single-phase pressure drop. This is achieved by proper design of the inlet throttling orifice at the core support plate and the mechanical fuel design including bottom plate, spacers and top plate.

Core stability simulations have been performed for the reference equilibrium cycle and the operational power/flow map shown in Figure 4.5-1. The predictions show that the reactor is stable with a decay ratio of oscillations always less than 0,6.

4.5.2.2.5 Control rod drives

The control rod drives (CRDs) utilize two separate drive mechanisms, one electro-mechanical and one hydraulic. The former is used for normal, continuous fine motion of the control rod - for burnup compensation or for adjustment of the power distribution - whereas the latter is used for rapid control rod insertion (scram). There are in total 213 control rods, divided into 16 scram groups; each group is equipped with its own scram module, consisting of a scram tank, piping and valve. The rods belonging to any one group are distributed over the core in such a way that the reactivity interference between them is virtually negligible. The consequence of a failure in one scram group is therefore no more serious than sticking of a single rod. The scram signal also initiates a rapid run-back of the recirculation pumps and a continuous insertion of all rods by the electro-mechanical drives, as a back-up to the hydraulic insertion.

4.5.2.3 Fuel handling and transfer systems

The fuel pools are located in the reactor service room on top of the containment. During plant operation, the reactor pool is filled with water to provide adequate radiation shielding. Spent fuel assemblies are brought up to the reactor pool and transported to storage racks in an adjacent fuel storage pool via a transport gate; in typical 12 month operation cycles 15-20% of the fuel assemblies are replaced during the refuelling.

Spent fuel is stored in the spent fuel pool for some years before it is transported away to an "away from reactor" storage facility in special transport containers, via the main transport shaft of the reactor building. The fuel pools are typically provided with storage racks that have sufficient capacity for storage of one complete core plus the spent fuel arising from seven to ten years of normal operation.

4.5.2.4 Primary components

4.5.2.4.1 Reactor pressure vessel

The reactor pressure vessel (RPV) is fabricated of carbon steel with an internal cladding of stainless steel. The general vessel arrangement (cf. Figure 4.5-2.) is the same as in the Forsmark 3 and Oskarshamn 3 plants; with steam and feedwater lines connected to the upper portion of the vessel and with the recirculation pump motor housings integrated with the pressure vessel at the lower portion.

Compared with the BWR 90, two major changes in the vessel design are easily distinguishable; the vessel diameter has been increased due to the increase in core power and size, and the number of motor housings for recirculation pumps has increased from eight to twelve. The reactor vessel height is approximately 21.5 m and the diameter is approximately 7.5 m.

4.5.2.4.2 Reactor internals

The shroud and the core support plate arrangement is closely similar to that of the BWR 75 design. All reactor internals are easily removable and replaceable. A visible difference refers to the lack of core spray nozzles; in the BWR 90+ the emergency core coolant injection is accomplished by a flooder system. The 205 steam separators, which new design has been developed in recent years, have a more open and simplified design compared to the old separator design. The improved separator units are more efficient, reducing steam moisture content, and have a reduced pressure drop.

4.5.2.4.3 Reactor recirculation pumps

The recirculation system consists of 12 internal glandless pumps driven by wet, asynchronous motors, supplied individually with "variable frequency - variable voltage" power from frequency converters. This type of pumps has been operating reliably in Westinghouse BWR reactors (for more than four million operating hours) since 1978. The 12 pumps provide a significant pump over-capacity, and this will contribute to enhanced controllability and capability of spectrum shift operation at the end of the operating cycle.

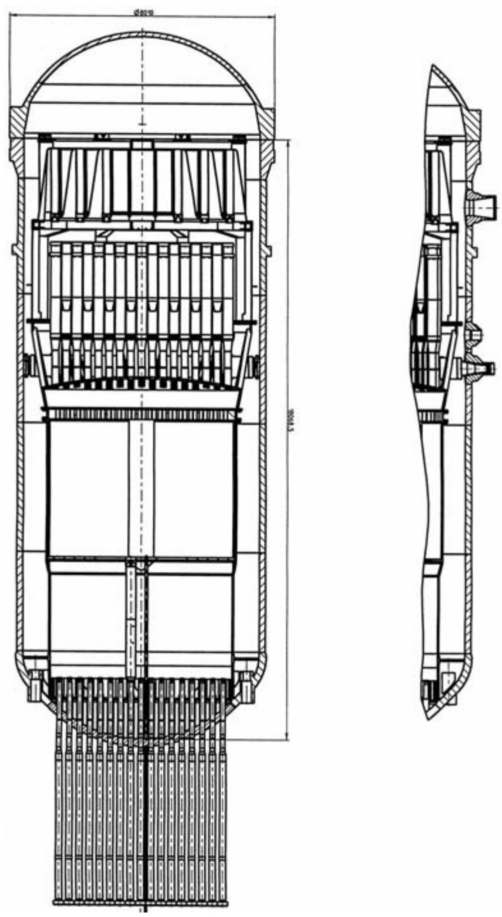

FIG. 4.5-2. Reactor pressure vessel and internals

4.5.2.5 Reactor auxiliary systems

4.5.2.5.1 Introduction

There is a large number of auxiliary systems in the nuclear power plant, so-called "balance of plant systems", systems that serve to cool and clean the water in the primary system and the water in the reactor service and spent fuel storage pools, ventilation systems etc. This section briefly describes some of the most important auxiliary systems.

4.5.2.5.2 Reactor water cleanup system

A main development objective for BWR 90+ related to the auxiliary systems is to evaluate possible design simplifications in order to achieve cost reductions and more straight-forward operation. In BWR 90+, the reactor water cleanup unit (RWCU) operation is controlled by the water chemistry in the reactor. During normal full power operation, clean-up needs are modest and only a small reactor water flow is passed through the RWCU, but whenever measurements show a need, the RWCU is taken into operation at higher capacity, as required. This arrangement reduces heat losses etc., and therefore yields cost reductions.

The RWCU in BWR 90+ comprises a radial flow type of deep-bed filters and heat exchangers (one of regenerative type). It takes water from the shutdown cooling system pumps and returns it partly as purge flows through the control rod drives and partly via the shutdown cooling system back into the vessel.

4.5.2.5.3 Isolation condenser

New safety requirements demand diversity and provisions against loss of the final heat sink normally used. To this end, a passive heat removal system, an isolation condenser, is incorporated in the BWR 90+ design. The design is quite similar to that of the isolation condenser in the first BWR built by Westinghouse (former ASEA- Atom), Oskarshamn 1. The design is well proven; it has attained more than 25 years of successful operation. The isolation condenser basically consists of two tube bundles in a pool of water. The pool is located at the same level as the fuel storage pools. The bundles are connected to the steam lines and the feedwater lines. During normal operation, a valve is closed on the condensate side. The system is initiated by opening this valve, and the system is driven by the density difference between the steam and the condensate, i.e. the design is passive.

The water in the pool will be heated. Later on, heat will be removed by boil-off from the pool. The evaporated water will be replaced by water from the water treatment and distribution system. As a backup, the system is connected to the reactor service pools.

The system will be actuated if the turbine condenser is isolated or if the steam line isolation valves close. The system will also be used during the initial phase during reactor cool down, ensuring cool down without any loss of primary water. After a short time, the system will control the pressure in the primary system, cool the core and remove the decay heat from the primary system.

4.5.2.5.4 Auxiliary feedwater system

During normal operation, the reactor is supplied with water at full pressure by the feedwater system. A separate auxiliary feedwater system is used during start-up and shut down when the need for feedwater is low.

The system is fed from the turbine condenser but can also use the fresh water storage tank as a water source. Since the system can use the water in the turbine condenser it can be used during hot stand-by and during the initial phase of a start up.

The system is supplied with stand-by power supply from the priority non-safety gas turbine-backed grid. It consists of one high-pressure pump with a capacity of approximately 50 kg/s. The system serves as backup for the high-pressure core flooder system and the isolation condenser in order to ensure an adequate supply of makeup coolant to the RPV (cf. Figure 4.5-7).

4.5.2.6 Operating characteristics

A schematic overview of the interrelationship between the reactor control systems with respect to operation is shown in Figure 4.5-3.

The internal recirculation pumps provide means for rapid and accurate power control and are advantageous also for load following purposes. The BWR 90+ plant is characterized by the capability to accept a 10% step change in power with an equivalent time constant of down to 5 seconds, and ramp load changes of 20% per minute is accepted. In the high power range, between 70 and 100% of nominal power, daily variations with the above change rate can be accommodated without restrictions; for wider power variations, the extended range is achieved by control rod pattern adjustments - at a rate of change of 1-2% per minute. Daily load following in a 100-40-100% cycle with (1 -) 2 hour ramps can be accommodated.

With respect to operating characteristics it may be noted that the plant is designed to withstand a full load rejection without being tripped; the plant will shift to house load operation, being prepared for a return to normal operation. The plant is further designed to avoid a reactor trip in the event of turbine trips, as long as the turbine condenser remains available for steam dumping, and to withstand certain grid voltage disturbances (voltage drops due to short circuits and other electrical faults on the grid) without being disconnected from the grid.

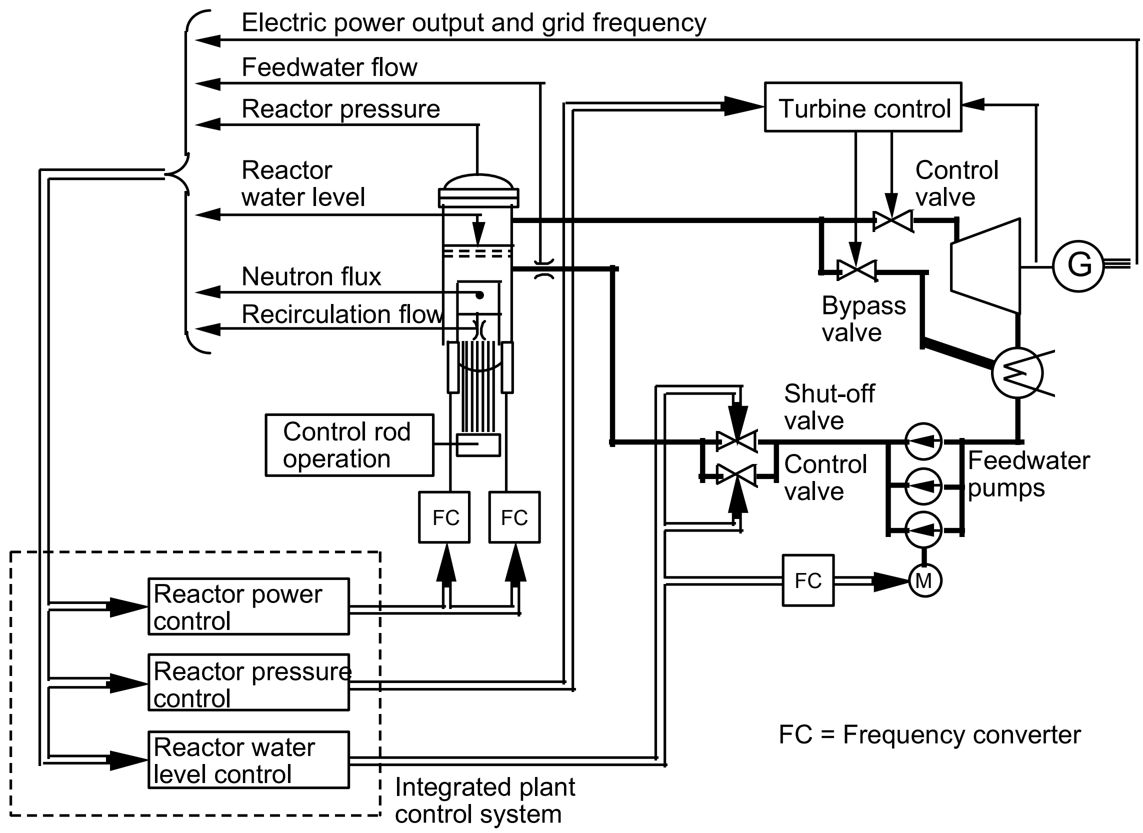

FIG. 4.5-3. BWR 90+ Interrelationship between reactor control systems

4.5.3 Description of turbine generator plant systems

The BWR 90+ design was developed by Westinghouse, Sweden, originally in collaboration with the turbine supplier, Alstom Power, Sweden. The reference turbine plant was based on ALSTOM's design for BWR applications which is well integrated with the BWR 90+ reactor and auxiliary systems, as well as the service and personnel buildings to achieve a highly functional nuclear power station unit with high availability, low specific costs and the best possible efficiency. The description below was based on the ALSTOM design for BWRs.

4.5.3.1 Turbine generator plant

The net power output will be 1550-1600 MWe, depending on the site conditions, in particular with respect to the temperature of the circulating water. The saturated steam from the reactor vessel is conveyed to the admission valves of the high pressure section of the turbine HP/IP cylinder via the four steam lines. After expansion through the HP section, the steam passes through a steam moisture separator unit and a 2-stage steam reheater, on its way to the admission valves of the turbine IP section and the three low pressure cylinders. A 110% capacity steam bypass system is also provided to enable dumping the full nominal steam flow directly to the main turbine condenser in the event of certain disturbances, in order to avoid pressure surges, and corresponding power peaks, in the reactor.

The generator is a four pole type turbo generator, designed for continuous operation with hydrogen as the cooling medium for the rotor and water as the cooling medium for the stator windings. Its rotor is directly coupled to the turbine. The electric power is transmitted to the three single-pole main transformers via individual, isolated air-cooled generator buses and to the external grid through HV circuit breakers. The exhaust from the low pressure turbine cylinders flows to the main turbine condenser located under the exhaust hoods of the low pressure turbine cylinders. The condenser is designed to accept also the steam flow from the main steam bypass system on start-up, hot standby and turbine trip. The condenser is cooled by the circulating water system which typically incorporates four electrically driven pumps; loss of one pump may call for a power reduction, but will not yield a turbine trip in the short term.

4.5.3.2 Condensate and feedwater systems

The condensate is pumped forward through the gland steam condenser, steam jet air ejector condensers, condensate cleanup by means of three 50% condensate pumps. The condensate is then pumped to the deaerator (or the feedwater tank) through the low pressure heaters by means of three 50% condensate booster pumps. The drainage from the low pressure heaters is directed via the condenser through the condensate clean-up system, to allowing about 35% of the total condensate flow bypassing the condensate cleanup system.

The feedwater system consists of the main feed pumps, two high pressure feedwater heaters, and associated piping. There are four 33% adjustable speed, electrical motor driven main feed pumps, drawing from the deaerator (the feedwater tank). Drainage from the high pressure heaters is routed to the deaerator. The power supplies to the FW pumps are utilizing static converter units which eliminate the large inrush currents at direct on line starting and therefore reduces the requirements on "voltage stability" (or rather short circuit strength) of the auxiliary power supply system busbars. Feedwater flow control is achieved by adjusting the feed pump speed. Extraction steam for the deaerator and high pressure heaters is provided from the high and intermediate pressure turbine extraction points, including moisture separator drainage and steam reheater exhaust. Extraction steam to the low pressure heaters are supplied from the low pressure turbines.

4.5.3.3 Auxiliary systems

The condensate clean-up system is composed of precoat filter units in order to remove impurities in ionic and particle form from the condensate. There is also an offgas system for treatment (delay and filtering) of potentially radioactive gases before releases to the atmosphere.

4.5.4 Instrumentation and control systems

4.5.4.1 Design concept, including control room

Modern process control and communication technology is applied to the BWR 90+. Its control and instrumentation systems are mainly based on micro-computers.

Utilisation of hardware modules and basic software from a standard industrial digital system series that is available on the market minimises maintenance and the necessary stock of spare parts.

The design of the I&C is determined by the safety classification of the equipment. I&C equipment with major safety importance (e.g., the Reactor Protection System) is classified as safety class 2 [category A according to IEC 1226], whereas other less important equipment is classified as safety class 3, 4 [or category B, C] or other equipment.

The major portion of the plant systems is supervised and operated from the central control room both during normal plant operation and accidents. The work in the control room is largely carried out at work positions, such as:

- Reactor safety system operating desk,
- Reactor operation system desk,
- Turbine system desk,
- Service system desk (including systems such as radwaste, ventilation, service air and water etc.),
- One desk with information for the shift supervisor.

The work positions are provided with Video Display Units (VDUs). The information needed for the actions, which are to be taken at that work position, is available at the VDUs. This arrangement is supplemented with an overview panel, on which an "overview" of plant functions and status is provided by conventional instruments as well as computer-based, large VDU displays (VDU projections or EL displays), as indicated in Figure 4.5-4.

The overview presentation shows the main process in the form of a flow diagram and indicates the status (normal, disturbed or failed) of various plant functions in correspondence with the operating instructions for the plant. It is visible to all operators in the control room.

The alarm display system will alert the control room personnel when a fault is indicated in a process system or when important plant process parameters have passed pre-set specified limits.

The status of safety systems and functions is presented in a similar way. The parameters that are of immediate interest in a disturbance situation are presented in a direct form. This means that the reactor pressure vessel with in- and outflow connections, together with neutron flux, water level, and reactor pressure, as well as control rods fully in (or not), are displayed directly on the overview panel.

4.5.4.2 Reactor protection and other safety systems

The reactor protection system (RPS) and the other safety control systems are built in a four-division configuration; process monitoring, signal treatment and conditioning take place in four independent channels (or divisions). Trip functions are generally generated in two-out-of-four coincidence logic in each individual ”outgoing” division for all RPS functions.

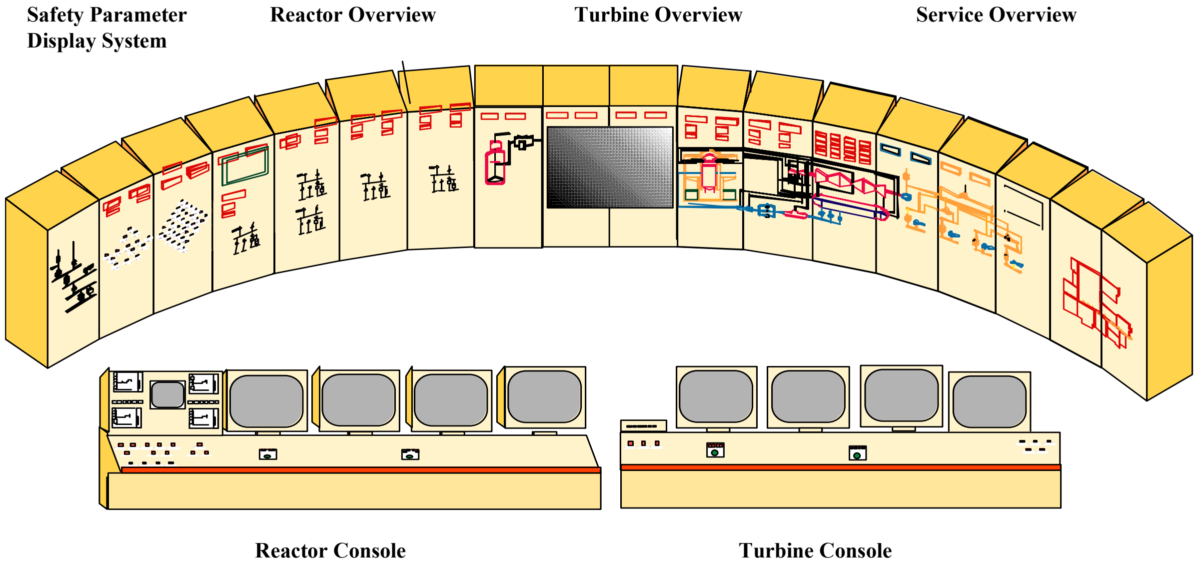

FIG. 4.5-4. Main control room arrangement

The reactor protection system collects signals from monitors and measuring transducers, which supervise a multitude of parameters important to the safety of the plant. Such parameters include e.g. neutron flux in the reactor core, reactor pressure, and reactor water level. The incoming signals are processed in a logic unit, and whenever required, adequate control signals are transmitted to initiate appropriate safety measures. The system design is such, that no single failure within the system will affect the intended safety function of the system.

It should be noted that the digital RPS is diversified and may include hardwired as well as diverse digital technology.

4.5.5 Electrical systems

The basic single line diagram for the electric power systems of BWR 90+ is shown in Figure 4.5-5.

The ratings of some major plant loads have been reduced by design changes in process systems. Modern switchgear components, having higher short circuit current ratings, have also become available, and consequently a significant simplification of the structure of the auxiliary power supply systems has been made possible. Another visible feature is the simplification at the DC distribution level; DC distributions at several voltage levels for power supply to various types of control equipment have been replaced by power supply from battery-backed AC distributions, using distributed AC/DC converters for the supply to the different types of equipment. On the other hand, it may be observed that there are now four separate battery-backed systems for power supply to non-safety systems. This arrangement contributes significantly to an improved separation – and functional independence - between safety and non-safety equipment and systems.

4.5.5.1 Operational power supply systems

Under normal operating conditions, the auxiliary electric power consumed within the plant is supplied from the general 10 kV system. It is fed either from the generators or from the 400 kV grid through the main transformer and the two plant transformers. If neither of these two supplies is available the general 10 kV system is fed from a 110 kV start-up grid through a start-up transformer.

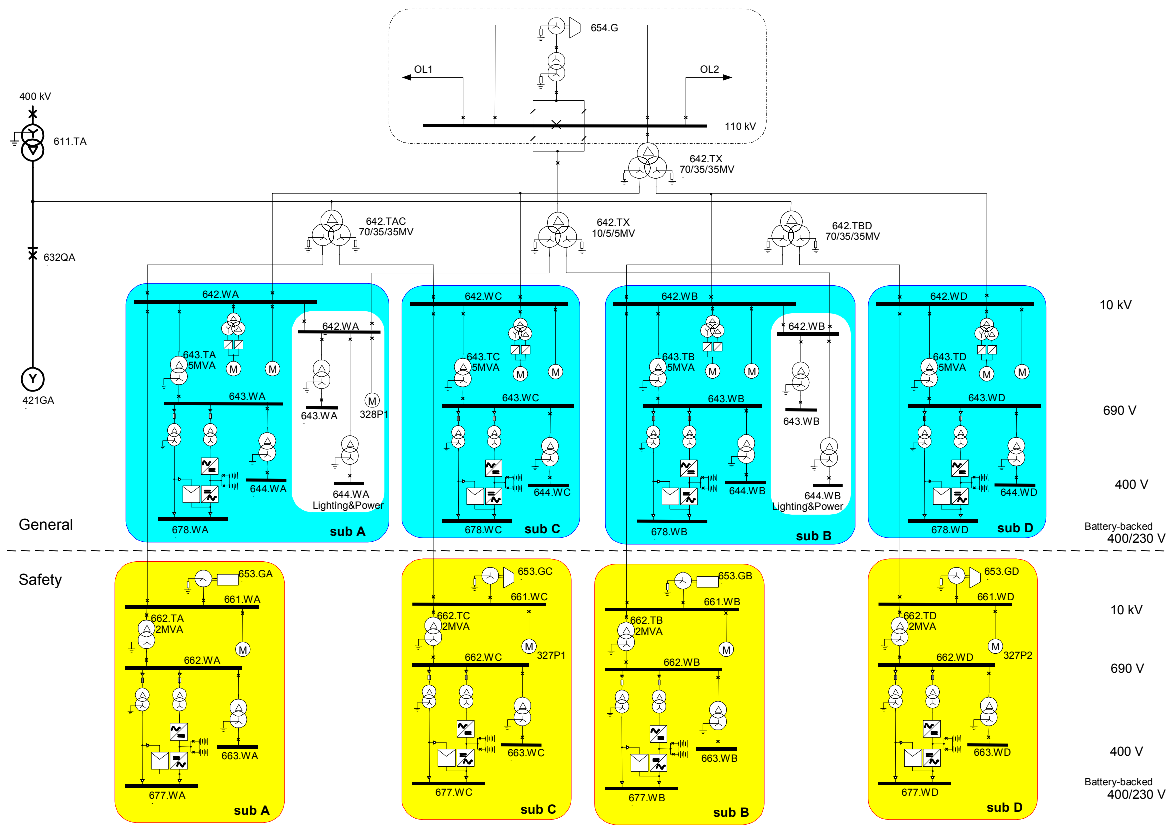

FIG. 4.5-5. BWR 90+ - Single line diagram

The general 10 kV system consists of six 10 kV busbars, of which four supply power to pump driving equipment, from the plant transformers or from the start-up transformer. These 10 kV system busbars feed transformers for the 690 and 400 V busbars of the general, non-safety-related systems as well as the remaining two 10 kV busbars aimed for priority demands. In addition the four general 10 kV system feed the safety gas turbine and diesel-backed 10 kV busbars.

The start up grid can in addition to the outer supply be fed from a gas turbine generator for priority non-safety power supply. The general 10 kV grids for priority power supply is fed from the start up grid through a priority transformer.

The start-up grid is connected to the two priority 10 kV busbars via a transformer that is provided also with a feeder from a gas turbine for priority non-safety power supply; a sectioning breaker is provided between these two incoming feeders. The pump motor for the auxiliary feedwater system, is supplied from the gas-turbine-backed busbar, and so are transformers for 400 V power to lighting and power outlets as well as backup supplies for other 690 V busbars.

There are four separate battery-backed 230 V AC systems which are fed by the general 690 V busbar system or from batteries via DC/AC converters. The batteries are fed from the general 690 V grid via transformers and AC/DC converters.

4.5.5.2 Safety-related systems

The electrical power systems for safety objects are strictly divided into four independent and physically separated sub-divisions - a principle that was implemented in the operating BWR 75 plants and maintained in the BWR 90+. According to this principle, the safety systems, electrical as well as mechanical systems, are designed with four sub-systems, in such a way that the function of any two of the sub-divisions will be sufficient to cope with design basis accidents. In other words, the four sub-systems principle implies that the plant can withstand a single failure even if one sub-division or component is not operable due to repair or maintenance.

Hence, there are four sets of diesel- or gas turbine-backed busbar systems, two diesel generators and two turbine generators. There are four AC/DC-DC/AC converters with intermediate batteries, and four battery-backed AC busbar systems - all without interconnections. The introduction of gas turbine generators for ensuring the power supply to two of the four safety system busbars was made possible by increased margins in the reactor system, and this diversification – by having diesels and gas turbines, - reduces the probability of total loss of AC power.

The diesel-backed systems feed electric power to systems and components necessary for the safe shutdown of the reactor, core cooling and the transport of residual heat to the ultimate heat sink. The systems consist of four physically and electrically separated busbars, which supply the pump driving equipment of the emergency core cooling systems (the high-pressure and the low-pressure coolant injection system), the transformers of the 400 V diesel-backed low voltage system, and the battery-backed AC systems. The diesel-backed systems are normally supplied from the general 10 kV system, but in the event of a failure in this system, the busbars will be powered from their stand-by power source, the diesel generator and turbine generator units, respectively. The switch-over to supply from the stand-by power plant (diesel generator or gas turbine generator) is initiated by the stand-by power plant control equipment.

The battery-backed 400/230 V AC system is supplied with power from the diesel-/gas turbine-backed busbars through DC rectifiers to batteries and through AC inverters to the busbars. The batteries provide uninterrupted power for two hours if the diesel-/gas turbine-backed busbars should not be available.

4.5.6 Safety concept

4.5.6.1 Safety requirements and design philosophy

The safety system configuration in the BWR 90+ design is characterised by a mixture of diversification, redundancy and separation, including the use of passive systems. The four-train principle with independence and separation, originally introduced in the BWR 75 design, will be maintained. This principle has yielded cost savings in the field of operation and maintenance since it provides possibilities for inspection, testing and maintenance activities during normal operation.

Apart from the inherent safety features of a boiling water reactor, the plant is protected by the reactor protection system and by engineered safety systems. These systems are actuated automatically and prevent erroneous plant operation or equipment malfunctions from causing a hazardous situation. They also minimise the amount of radioactive material released upon postulated accidents. The degree of automation is based on the assumption that no manual action by the operator(s) shall be needed during at least 30 minutes following the occurrence of an accident, with respect to protection of the core and prevention of radioactive releases to the environment.

The elimination of large-diameter pipe connections to the bottom region of the RPV results in a limited need for rapid injection of water to the core in the event of a large LOCA. Therefore, it is possible to introduce diversified means, - diesel generators and gas turbine generators, - for the power supply to the pumps of the flooder system; the longer start-up time of the gas turbine of about 1 minute, compared to about 10 seconds for the diesel generators, is fully acceptable.

The most important tasks of the safety arrangements following an accident are:

- to shut down the reactor
- to isolate the reactor containment
- to provide emergency core cooling
- to remove the residual heat
- to mitigate the consequences of the accident.

4.5.6.1.1 Redundancy and separation principles

Safety systems in BWR 90+ are designed to comply with the single failure criterion. The safety systems are divided into four redundant subsystems or "trains". In transient and postulated accident situations, one or two of the four subsystems will be sufficient to accomplish the intended safety function.

In the reactor building, the four subsystems are located in separate bays ("H bays"), one in each quadrant, adjacent to the reactor containment and surrounded by thick concrete walls. The physical separation is maintained all the way to the ultimate heat sink.

Separation and independence criteria for electric equipment are also given by the IEC 709 standard. The main control room is functionally separated from the I&C equipment in the four safety trains. Safety I&C systems are arranged in four channels with actuation based on a “two-out-of-four” channel logic.

Figure 4.5-6. shows the most important safety equipment in the four safety trains. The two standby power diesel generators with their ancillaries are installed in diesel buildings A and B, and the two standby power gas turbines with their ancillaries in diesel buildings C and D. The A/C and B/D buildings are located at opposite sides of the reactor building; this provides a high degree of physical protection with respect to external impacts. These buildings also house auxiliary power supply and control equipment with safety function, as well as pumps and heat exchangers for cooling systems with safety function.

The consistent implementation of the separation principles leads to a design that is extremely well protected against failures caused by "local" events, such as a fire or sabotage.

4.5.6.1.2 Diversity

Probabilistic safety analyses, which represent a powerful tool for evaluating system configurations, often show that a suitable diversification is favourable with respect to total functional reliability by lowering the probability for common-cause failures. Therefore, a high degree of diversity, including passive functions, has been sought for safety system functions in the BWR 90+ design. This includes safety functions such as reactor shut down, core cooling, decay heat removal, and emergency power supply. The following are main examples:

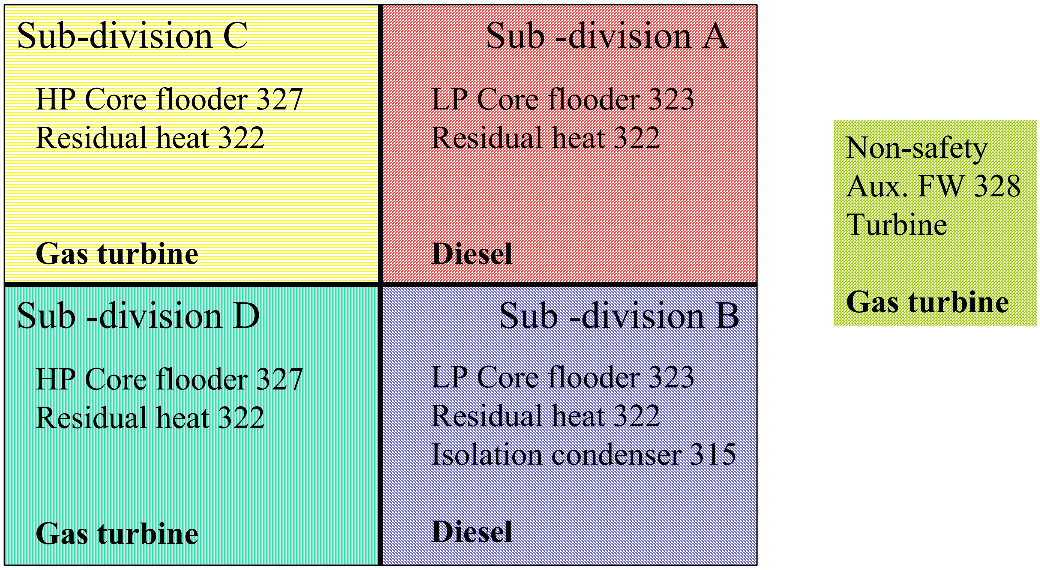

FIG. 4.5-6. The four sub-divisions of safety systems

- Diverse control rod insertion: Hydraulic scram and electric motor-operated insertion,
- Diverse means for reactor shutdown: Control rods insertion and a passive boron system injection,
- Diverse design for half of the 16 safety/relief valves, as well as diverse actuation by electric drives and pneumatic pilot-operated valves,
- Diverse valves in the ADS function,
- Diverse emergency core cooling: achieved by high-pressure core flooders for emergency core cooling and LOCA and automatic depressurisation (ADS) together with the low-pressure core flooders, with the isolation condenser and the auxiliary feedwater system as backup;
- Diverse emergency power supply with diesels and small gas turbines,
- Diverse residual heat removal: This function can be achieved by the shutdown cooling system or by the emergency cooling systems, backed by the isolation condenser.
- Diverse heat sink: heat removal is accomplished by the priority cooling water system to the inlet or outlet water channels. In addition, the isolation condenser provides capability of reactor pressure control, core cooling and decay heat removal, as backup;
- Diverse power supply by diesel generators and small gas turbine generators to the residual heat removal system, and in addition, a separate gas turbine as backup;
- Diverse operation of containment isolation valves: The inner containment isolation valves are preferably check valves or pneumatically operated valves, most of the outer valves are operated by electric motors.
- A mobile and diverse power supply unit for containment spray cooling system after a severe accident.

Some examples on passive system functions in the BWR 90+ are:

- the passive boron system;
- the isolation condenser;
- the scram insertion of the control rods; and
- the containment function with a passive core catcher in the event of a severe accident.

In addition, it should be noted that the containment has been designed in such a way that the possibility of core uncovery during refuelling outages has been eliminated.

4.5.6.2 Safety systems and features (active, passive and inherent)

4.5.6.2.1 Hydraulic scram system

The control rods are divided into 16 scram groups each comprising 10 to 14 rods. Each group is equipped with its own scram module, consisting of a scram tank at high pressure, piping and actuating valve. The different control rods of a scram group are sufficiently separated to achieve a negligible reactivity coupling between them. The reactivity consequence of a failure in one scram group is therefore no more serious than sticking of a single rod for the cold, clean shutdown core, and the reactor will be sub-critical in all situations even if a single failure should occur in one of the scram groups. The scram signal also initiates a rapid run-back of the recirculation pumps and a continuous insertion of all rods by the electro-mechanical drives, as a backup to the hydraulic insertion.

4.5.6.2.2 Control rod operating system

The electrical motors of the electro-mechanical control rod drive system are normally used to adjust the control rod pattern during start-up, normal shutdown and under normal operating conditions. The electro-mechanical system is capable of fully inserting all control rods into the reactor core in less than 4 minutes after an initiating event.

The diversified means of control rod actuation and insertion (together with a generous reactor pressure relief capacity) provide, in combination with a capability of rapid reduction in the recirculation flow rate (recirculation pump run-back), a countermeasure against ATWS (anticipated transients without scram).

4.5.6.2.3 Boron system

Negative reactivity can be inserted into the reactor core by injection of a boron solution. The system function is actuated manually or automatically as a substitute for the control rods, and it is capable of keeping the reactor sub-critical at cold conditions with no control rods inserted. BWR 90+ incorporates an improved boron system. The system consists of vessels containing boron solution. The vessels are connected to gas accumulators containing nitrogen at high pressure. The gas accumulators are located outside the reactor building. Each boron vessel is connected to the RPV. When the system is actuated, shut-off valves are opened and the boron solution will be injected into the reactor.

4.5.6.2.4 Relief system

The system consists of 16 safety/ relief valves connected to the main steam lines inside the reactor containment vessel. The safety (relief) valves are own-medium operated valves, each being controlled by two pilot valves, one pressure-activated and one electrically controlled. The steam released by the relief system is conveyed to the condensation pool through relief pipes.

The safety relief valves are divided into two diversified groups, each containing 8 valves. Four valves from each group will act as relief valves and four as safety valves.

The relief system can also be activated by the automatic depressurisation systems (ADS) of the reactor protection system (RPS) to depressurise the reactor in order to enable core cooling by the high capacity low-pressure coolant injection system, should such a need arise. The ADS function is also diversified with five valves from each group.

BWR 90+ is also equipped with 4 valves designed to blow water at high water levels in the core. These valves can be used as an ultimate backup to the safety relief function. They will open on an ADS signal, and will remain open after depressurisation of the reactor, ensuring that the reactor pressure will be kept low during a severe accident. BWR 90+ is provided with an isolation condenser, which following a first, short opening of the safety relief valves, will control the pressure during a transient.

4.5.6.2.5 *Emergency cooling systems*

Core flooder systems

The emergency cooling systems are divided into two groups:

One group comprising two sub-systems, each with a 100 % capacity pump that is capable of cooling the core at full pressure. In addition, the pumps have a low-pressure capability and can act as low-pressure flooders. The low-pressure cooling capability is 50 % for each pump.

The other group, also with two pumps, contains only low-pressure core flooders with a capability of 50 % for each pump.

The arrangement of the core cooling systems is outlined in Figure 4.5-7.

The isolation condenser and the auxiliary feedwater system will serve as a backup for the high-pressure flooder system; both these systems have a cooling capacity of 100 %. These systems were described in more detail earlier.

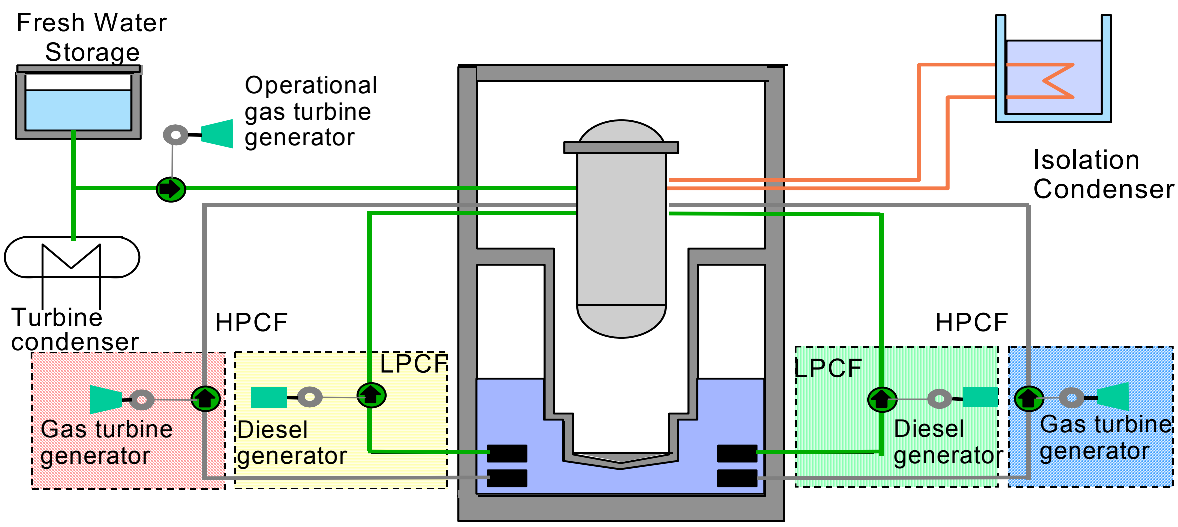

FIG. 4.5-7. Core coolant make-up systems

Residual heat removal system

This system, together with the shutdown secondary cooling system and the shutdown water cooling system, constitutes the emergency residual heat removal chain to the ultimate heat sink. Water from the containment condensation pool is circulated through a heat exchanger and is discharged into the containment wetwell through a system of spray nozzles located above the condensation pool. After a loss-of-coolant accident, the spray may manually be switched to the drywell. The cooling function has been strengthened by a switch-over capability to the cooling water outlet channel, should the cooling water intake become blocked. The residual heat removal system is outlined in Figure 4.5-8.

The isolation condenser serves as a diversified ultimate heat sink and as a backup for the residual heat removal system. The containment vessel spray will be used as the primary system for cooling the containment after a severe accident. In order to increase the functional reliability, the system is diversified into two subsystems with diversified components and power supply, gas turbines and diesels. In order to improve the reliability, the system has an alternate power supply; each pump in the system can be connected to a mobile power supply unit to guarantee the function even in case of a total blackout. As a backup to these functions the containment spray system can be used to fill the containment with water and will then, together with the FILTRA/MVSS-system cool the containment (cf. Figure 4.5-11).

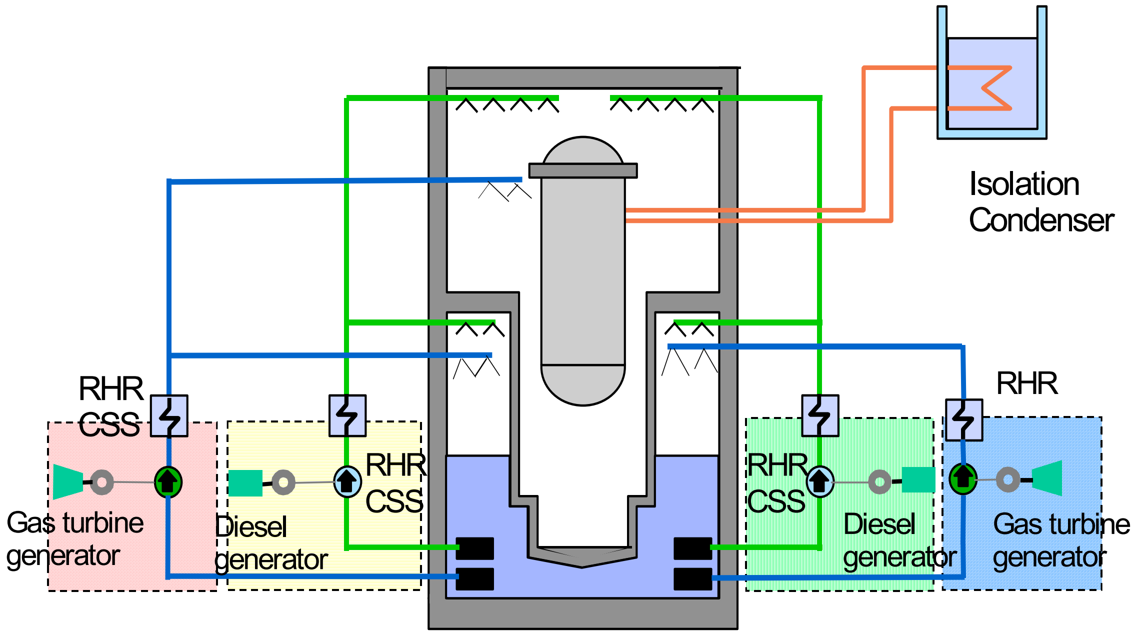

FIG. 4.5-8. Residual heat removal systems

4.5.6.3 Severe accidents (beyond design basis accidents)

The Severe Accident Management Strategy (SAMS) aims at preventing unacceptable environmental consequences and to provide a final stable plant state - even in a severe accident that involves severe damage to the reactor core. This is accomplished by ensuring an extremely low frequency for sequences that involve a loss of containment function prior to severe core damage, and by protecting the containment properly against the impacts of the different severe accident phenomena. The containment integrity in a severe accident situation is ensured by mitigative design solutions, that range from e.g. reactor system depressurisation to avoid core melt and reactor pressure vessel melt-through at high pressure, to arresting molten core material in the core catcher arrangement in the containment.

The top-level design targets applied to SAMS with respect to the effectiveness and reliability of the various plant design features, are the following:

- There shall be no need for manual actions during the first 12 hours from accident initiation.
- There shall be no need to discharge gases out of the containment during the first 24 hours from accident initiation, even if assuming100 % oxidation of all core zirconium.
- The target value for releases after a severe accident has been defined in international regulations (e.g. Finnish regulations and YVL Guides. The frequency of severe accident sequences and scenarios that could lead to exceeding the target value shall be below $5 \cdot 10^{-7}$/RY (mean).
- The monitoring and control functions shall have sufficient coverage that the SAMS functions can be reliably activated, their effect well surveyed, and also the general plant status after a severe accident derived from the measurements. The SAMS delineates necessary severe accident instrumentation and constitutes the basis for developing appropriate operator procedures.

The BWR 90+ design fulfils all these design targets.

4.5.7 Plant layout

4.5.7.1 Buildings and structures, including plot plan

The general arrangement of the buildings is depicted in Figure 4.5-9. It is characterised by a division into an essentially nuclear and safety portion, consisting of the reactor building and the diesel buildings, and a more conventional portion that comprises the turbine generators and auxiliary systems of the plant. The "conventional" part is physically separated from the former. This arrangement is advantageous when constructing the plant as well as during plant operation, since the conventional part does not interfere significantly with the nuclear part.

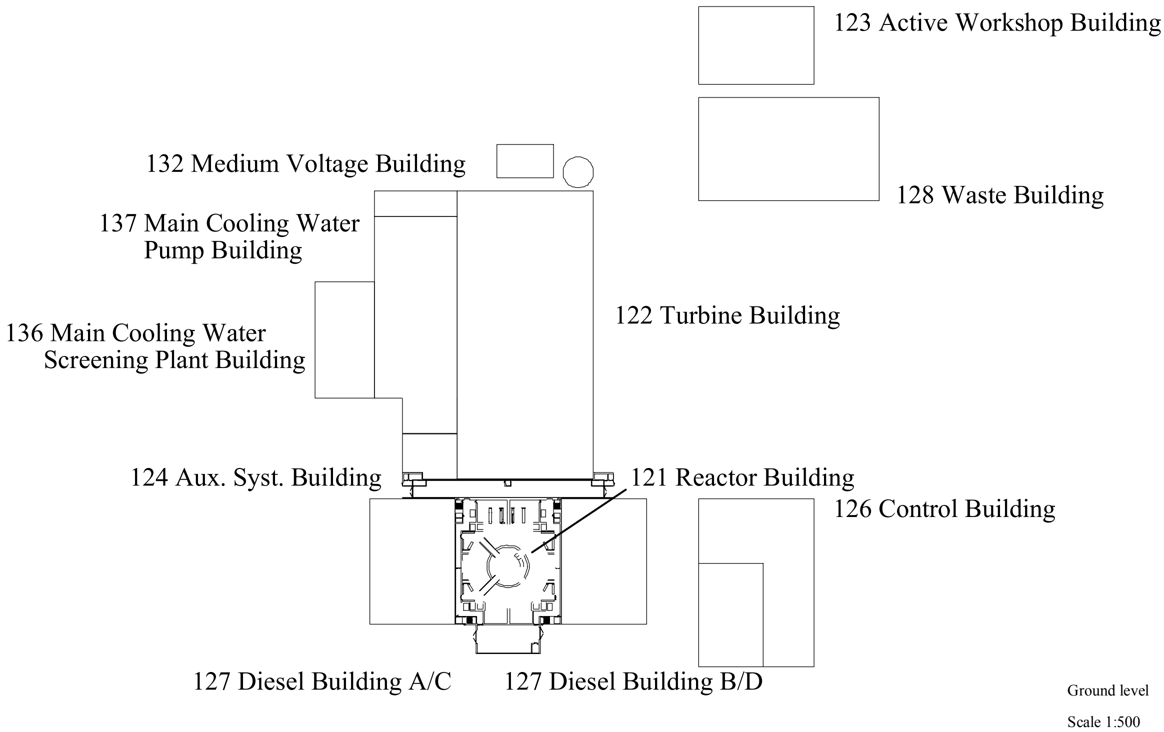

FIG. 4.5-9. General building arrangement

4.5.7.2 Reactor building

The reactor building encloses the primary containment completely and is designed to serve as a secondary containment, kept at under-pressure by ventilation systems in which the exhaust air can be re-routed to filters when needed. The reactor building houses all primary process and service systems outside the primary containment, including handling equipment for fuel and main components, fuel pools, reactor water cleanup system and engineered safety systems. Engineered safety systems are located in the bottom part of the reactor building. The top of the reactor building serves as a reactor service room with pools for reactor service, for storage of internals during refuelling, and for storage of spent fuel and "failed" irradiated components, space for tools and handling equipment and cranes and platforms (refuelling and service bridges, overhead crane). During normal operation of the plant, the reactor building is kept at a certain pressure (typically 400 Pa) below ambient. Thus, any release of radioactive matter into the reactor building will be channelled to the 110 m high main stack.

4.5.7.3 Containment

The primary system, the reactor coolant pressure boundary, and important ancillary systems are enclosed in the primary containment, a cylindrical pre-stressed concrete structure with a steel liner which ensures leak-tightness. The containment also acts as a biological shield against radiation from the reactor.

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.

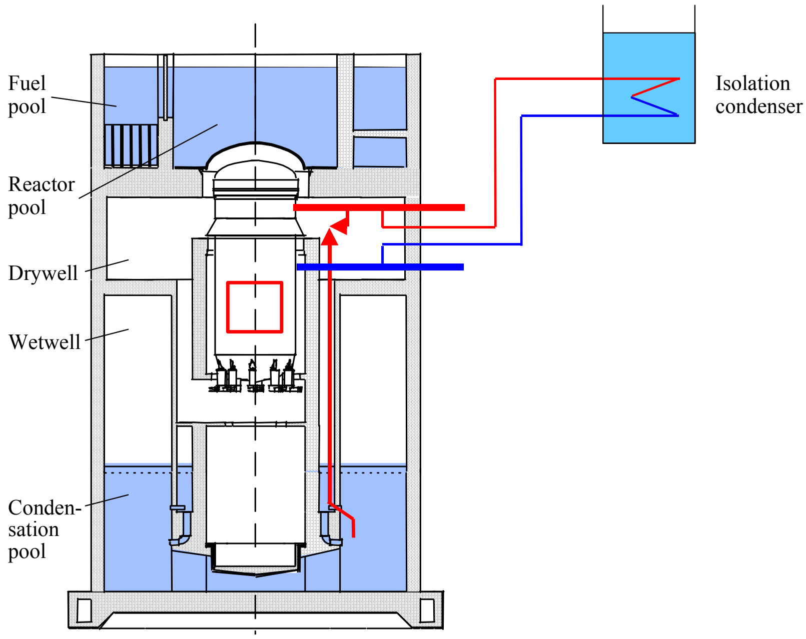

FIG. 4.5-10. Primary containment arrangement

The primary containment is of pressure-suppression type, with two major compartments - a drywell and a wetwell. 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 intermediate floor, is provided with a leak-tight liner in stainless steel. The relief pipes from the relief valves in the pressure relief system are routed through horizontal passages, leaving the partition floor without any penetrations; the probability of a degraded pressure suppression function has been practically eliminated. The containment arrangement is depicted in Figure 4.5-10.

During normal operation, the containment atmosphere is inerted by means of nitrogen gas, thereby eliminating the risk of fires during operation and the risk for hydrogen explosions in case of postulated core melt accidents. The wetwell gas compression chamber volume and the condensation pool water volumes have been increased compared with previous designs.

The design goals set for the BWR 90+ containment were to enable reducing the construction time and enhancing safety, particularly in the following areas:

- The review of BWR 90 against the EUR, as well as the new safety requirements in Finland, have demonstrated that the containment volumes must be sufficient to accommodate oxidisation of 100 % of all core zircaloy in the event of a severe accident, without need for filtered venting within at least 24 hours, and with no need for manual actions within at least 12 hours.
- PSA studies of shutdown conditions for the BWR 75 indicated that human errors during service of reactor coolant pressure boundary (RCPB) components could be a significant contributor to the frequency of large releases. To reduce this frequency, the containment shall have the ability to prevent uncovery of the core even in the event of such human errors.
- The pressure retaining structures of the containment shall not directly be exposed to molten core material in the event of core melt and RPV melt-through.
- To reduce the vulnerability to ex-vessel steam explosions by providing a robust design which could resist these loadings, but also reduce the potential for water in the lower drywell at RPV melt-through.
- To simplify demonstration of coolability of core debris.

The BWR 90+ containment fulfils all these design goals.

A dry core catcher (cf. Figure 4.5-11.) is located under the reactor pressure vessel; it is submerged into the containment pool. In case of a severe accident involving core meltdown and penetration of the reactor pressure vessel, the containment bottom slab and wall structures do not constitute the primary barrier against core debris. The molten core will be collected in the core catcher, and the water surrounding the catcher will provide cooling of the molten material. The improved containment design reduces the risk for steam explosions and a molten core will be cooled in a passive way by the containment pool water. In addition, the core catcher ensures that there is no risk of major core-concrete interaction.

With respect to arresting the melt in the core catcher, it may be noted that the core catcher may experience various types of transient loads before a final steady-state configuration is reached. The most demanding final state is the pool of core melt retained in the core catcher structure. The decay heat generated in the core melt is transferred by convection to the pool boundaries; its top surface will be directly cooled by water flooded into the lower drywell. The decay heat transferred to the core catcher structure is removed by the condensation pool water that forms a natural circulation loop outside the core catcher. Similar to the scheme of in-vessel melt retention, the core catcher structure will maintain its integrity due to the large cooling area and the low outer surface temperature. The core catcher is a passive device in itself.

The improved design will cope with the pressure build-up in a passive way for one day without activation of the overpressure protection, assuming hydrogen generation from all zirconium in the core. The containment itself serves as an inherently passive system ensuring that no releases of radioactivity to the ambient will occur during the first period after a severe accident with a molten core. Spraying water into the drywell, as well as, activating other active cooling systems, will cool the containment, transfer activity to the containment pool, reduce the containment pressure and prevent releases to the surroundings.

All nuclear power plant containment structures may be characterised as pressure vessels. Logically, and in compliance with the code requirements of pressure relief equipment for pressure vessels, all Nordic BWRs, as well as the BWR 90+ design, are equipped with pressure relief equipment for the containment. This consists 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 may then be released via a filtered vent system without concerns for significant off-site consequences.

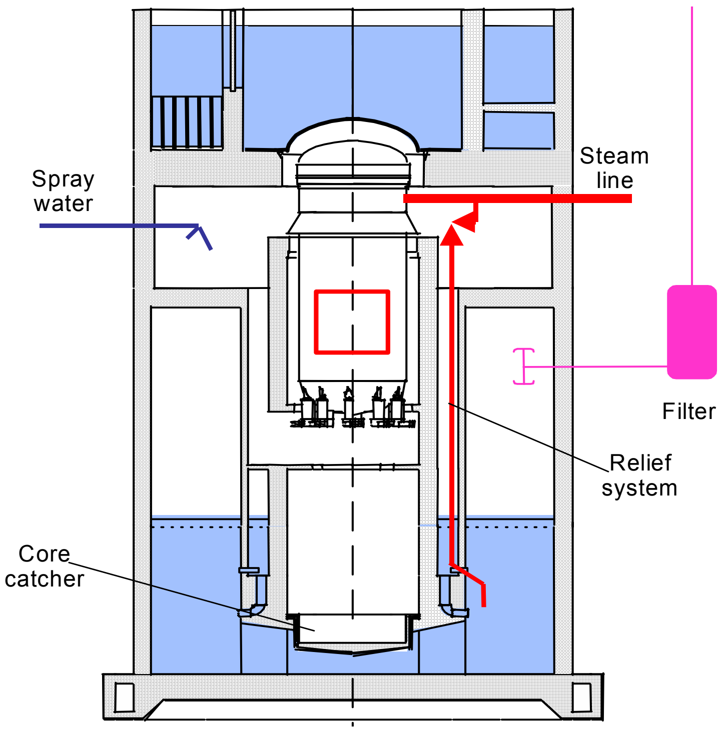

FIG. 4.5-11. Severe accident mitigation features

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 floor. In case of a LOCA induced by human errors during plant shut-down and refuelling operations, the water volume of the pools above the reactor will be sufficient to fill the containment drywell volume up to above the upper drywell (or partition) floor, consequently, this design implies that the core will be kept flooded without human actions or system actuations.

The improved containment design is fully adapted to the modular building technique and thus reduces the construction time and costs. The main features of the improved containment design are:

- Reduced construction time and costs;
- Minimised risk for drywell - wetwell bypass;
- Core remains covered by water if loss of coolant accident occurs during refuelling;
- Passive core melt retention and cooling inside containment; no releases within one day in the event of a core melt accident;
- Severe accidents have been taken into account at an early stage;
- Containment structures do not constitute primary barrier for core debris;
- A dry core catcher reduces the risk for steam explosions;
- Core concrete interaction is negligible;
- Increased volumes accommodate pressure build-up from hydrogen generation at a core melt accident;
- Nitrogen gas inerted atmosphere allows initiation of water spraying to cool structure and reduce pressure - without risk for hydrogen explosions;
- Ultimate containment over-pressure protection by filtered venting.

4.5.7.4 Turbine building

The turbine plant is accommodated in the turbine building. The turbine section, which occupies most of the building, contains the turbine unit feed heaters, moisture separators and reheaters. Radiation protection walls are provided at suitable points in order to improve the accessibility during normal operation. The turbine hall is equipped with overhead travelling crane for heavier lifting duties.

4.5.7.5 Other buildings

The so-called diesel buildings are located on both sides of the reactor building and structurally integrated with it to enhance the protection against the effects of earthquakes. The two buildings contain most of the equipment of the safety systems that is located outside the reactor building; they house the two divisions of diesel generators and the two divisions of gas turbine generators, the associated busbars of the diesel-/gas turbine-backed auxiliary power supply system, the AC/DC-DC/AC converters, batteries and busbars of the battery-backed power supply system, safety control equipment, pumps, valves and heat exchangers for the intermediate closed cooling systems, and pumps and valves for the service water system. The buildings basically consist of two parts, each containing one safety division. This layout gives a good protection against fire and external events.

4.5.8 Technical data

General plant data

Power plant output, net	1 550-1 600	MWe
Reactor thermal output	4 250	MWt
Power plant efficiency, net	>37	%
Cooling water temperature	5	°C

Nuclear steam supply system

Number of coolant loops	1	
Primary circuit volume	~510	m^3
Steam flow rate at nominal conditions	2 304	kg/s
Feedwater flow rate at nominal conditions	2 304	kg/s

Reactor coolant system

Primary coolant flow rate	14 300	kg/s
Reactor operating pressure	7.5	MPa
Steam temperature/pressure	287/7.3	°C/MPa
Feedwater temperature	215	°C
Core coolant inlet temperature	281	°C
Core coolant outlet temperature	289	°C
Mean temperature rise across core	8	°C

Reactor core

Active core height	3.71	m
Equivalent core diameter	5.16	m
Heat transfer surface in the core	9 066	m^2
Average linear heat rate	14.5	kW/m
Fuel weight	153	t U
Average fuel power density	27.8	kW/kg U
Average core power density	54.9	kW/l
Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	3.99	m
Number of fuel assemblies	872	
Number of fuel rods/assembly	4x((5x5)-1)	
Number of 1/3 part length rods/assembly	4	
Number of 2/3 part length rods/assembly	8	
Number of spacers	4x8	
Enrichment of reload fuel at equilibrium core	3.34Wt%	
Operating cycle length (fuel cycle length)	12	months
Average discharge burnup of fuel	50 000	MWd/t
Cladding tube material	annealed, recrystallised Zr 2	
Cladding tube wall thickness	0.6	mm
Outer diameter of fuel rods	9,84	mm
Fuel channel/box; material	Zr-4	
Uranium weight/assembly	175,6	kg
Active length of fuel rods	3.71	m
Burnable absorber, strategy/material	Gd_2O_3 mixed with fuel	
Number of control rods	213	
Absorber material	B_4C	
Drive mechanism	electro-mechanical	
Positioning rate	30	mm/s
Soluble neutron absorber	Boron	

Reactor pressure vessel

Inner diameter of cylindrical shell	7150	mm
Wall thickness of cylindrical shell	189 mm	
Total height, inside	21196	mm
Base material: cylindrical shell	low-alloy carbon steel	
RPV head	[to ASTM A533, grade B, ASTM A508, class 3, or equiv.]	
lining	stainless steel	
Design pressure/temperature	9.1/305	MPa/°C
Transport weight (lower part)	800	t
RPV head	130	t

Reactor recirculation pump

Type	Internal, glandless, centrifugal pump	
Number	12	
Design pressure/temperature	9.8/310	MPa/°C
Design mass flow rate (at operating conditions)	1 200	kg/s
Pump head	0.2	MPa
Rated power of pump motor (nominal flow rate)	590	kW
Pump casing material	as for RPV	

Pump speed (at rated conditions)		1 200	rpm

<u>Primary containment</u>

Type	Pressure-suppression	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	29.3/38	m
Design pressure/temperature (LOCA)	550/172	kPa/^{o}C
Design leakage rate	0.5	vol%/day
Is secondary containment provided?	Reactor building	

<u>Reactor auxiliary systems</u>

Reactor water cleanup,	capacity	2 x 46	kg/s
	filter type	deep-bed	
Residual heat removal,	at high pressure	2 x 85	kg/s
	at low pressure (100 ^{o}C)	2 x 18.5	MW
Coolant injection,	at high pressure	2 x 50	kg/s
	at low pressure	2 x 240 + 2 x 90	kg/s

<u>Power supply systems</u>

Main transformer,	rated voltage	24/400	kV
Plant transformers,	rated voltage	24/10/10	kV
	rated capacity	70/35/35	MVA
Start-up transformer	rated voltage	110/10/10	kV
	rated capacity	70/35/35	MVA
Medium voltage busbars (6 kV or 10 kV)		8	
Number of low voltage busbar systems		8	
Standby diesel generating units: number		4	
	rated power	2	MW
Number of diesel-backed busbar systems		16	
Voltage level of these		10 000/690/400/230	V AC
Number of DC distributions		0	
Number of battery-backed busbar systems		8	
Voltage level of these		400/230	V AC

4.5.9 Measures to enhance economy and maintainability

With respect to economy, the design work aimed 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 was: "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 were incorporated also in the BWR 90+ design.

Some specific design and performance goals of the BWR 90+ development project were originally:

- Plant nominal power output; 1500 Mwe,
- Construction time; less than 1500 days,
- Energy availability; higher than 90 %,
- 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 nominal plant size was increased from 1350 MWe for the BWR 90 design to 1500 MWe for the BWR 90+ design. Studies of plant efficiency assuming a specific turbine plant design has yielded the interval 1550-1600 MWe net output as realistic figures. Results from BWR fuel development, e.g. the SVEA Optima2, were fully incorporated in the design.

Interest during construction represents an important portion of the cost for a new nuclear power plant, and a shortened construction time will thus be beneficial. An improved containment design and the reactor service room layout allow extensive use of slip-forming and modular construction methods. The reactor building and the auxiliary buildings will be built with 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 at least similar results, and exceed the EUR requirement of 87 % over 40 years of operation. Besides, an efficient feedback of operating experience from TVO, and the Swedish utilities, bring improvements regarding operation and maintenance aspects.

TVO has demonstrated the feasibility of very short refuelling outages, down to ten days or less, 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 highly realistic.

4.5.10 Project status and planned schedule

The earlier design BWR 90 was completed in 1991. In 1997, this design was selected by the EUR Steering Committee to be the first BWR to be evaluated versus the European Utility Requirement (EUR) documents. The review was completed in 1998. An "EUR Volume 3 Subset for BWR 90" document was issued in 1999.

Westinghouse (the former ABB Atom) has continued its BWR development work with an "evolutionary" design called the BWR 90+, aimed at developing the BWR as a competitive option for the anticipated revival of the market for new nuclear plants, as well as feeding ideas and inputs to the modernisation efforts at operating plants. The development was performed by Westinghouse Atom in close co-operation with TVO and Swedish BWR operators.

The BWR 90+ design builds firmly on proven design, but considers and adopts new developments including new technology, such as digitised control equipment, and passive systems and functions, as well as features that yield improved severe accident mitigation.

A precise classification of the project status is not easily made; the status of the BWR 90+ project lies somewhere between completed basic design and completed detailed design. To be more specific, the plant design is complete, except for some minor items. Detailed design/engineering has not been made for all systems and components, and detailed design, or detailed specifications, for procurement of all materials, components, systems, package units, construction/erection services, etc. have not yet been completed.

With respect to licensing, it may be noted that the authorities in Sweden never take on licensibility reviews, until a utility files for a construction permit. Such a design certification process does not exist. Reference is therefore made to the comment above on the close relationship with the BWR 75 design, and to the licensing discussions that have taken place with STUK, the Finnish licensing authority, in conjunction with TVOs application for a “decision in principle” for the "Finland 5" project.

Construction of a BWR 90+ plant can build directly on the experience gained from previous projects. The construction activities have been analysed by Westinghouse and TVO with a team of civil engineering, installation and commissioning supervisory personnel that built and commissioned the Oskarshamn 3 Nuclear Power Plant in Sweden in 57 months from the first pouring of concrete to start commercial operation.

The resulting schedule for the BWR 90+ plant shows that the total construction time for the plant - from pouring of the first structural concrete to start commercial operation - will be less than 1500 days, i.e., the total construction time stipulated in the EUR and EPRI URD will be met.

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4.6 EPR (FRAMATOME ANP, FRANCE/GERMANY)

4.6.1 Introduction

The European Pressurized Water Reactor (EPR) is the designation for a development effort by Nuclear Power International and its parent companies, Framatome and Siemens, whereas the nuclear part of both companies have merged in the meantime into a joint company called Framatome ANP (*A*dvanced *N*uclear *P*ower) as an entity in the Areva group. The project was performed in co-operation with Electricité de France and German Utilities, aiming at achieving a new improved nuclear power plant design that will become an acceptable and attractive alternative for meeting energy demands in the future.

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, named EPR. The German and French utilities decided at that time to establish, together with other European utilities, specifications that would represent common utility views on the design and performance of future nuclear power plants. A first draft of the generic (non-design-specific) sections of these European Utility Requirements (EURs) was issued in April 1994, and an extended version B taking into account comments to the draft from utilities and vendors was issued in April 1996. Compliance of the EPR with the EUR was confirmed in 1999.

The following description of the EPR is based on the harmonization between the partners at the end of the basic design.

4.6.2 Description of the nuclear systems

4.6.2.1 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 volumes compared to the current designs.

In the RPV design, the free 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 in case of LOCA or minimizing the core uncovery depth if any.

At the same time the increase of this volume contributes to an improvement in the mitigation of accidents at shutdown conditions (in particular mid-loop operation), e.g. with loss of the rsidual heat removal system (RHR), by providing longer grace periods.

For the pressurizer a large volume both in terms of water and steam phase is provided in order to smoothen plant response to relevant operating transients and accidents.

The large 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 be > 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 of four discharge trains (one of them dedicated for severe accidents) are mounted to the top of the pressurizer in order to minimize the length of 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 (bleed and feed); and
- 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 and its severe consequences.

4.6.2.2 Reactor core and fuel design

The core contains 241 mechanically identically designed fuel assemblies; somewhat more than in currently operating units. Each fuel assembly consists of 265 fuel rods and 24 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 15.5 kW/m, allowing average batch burnups of up to 60 GWd/tU for a 1 year cycle. 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. As a rule, slow reactivity changes caused by changes of xenon concentration and bumup 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 elements application 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. A cross-section of the core, showing the location of the core instrumentation, is depicted in Figure 4.6-1.

4.6.2.3 Fuel handling and transfer systems

The spent fuel assemblies are transferred to the fuel pool located in the fuel building. The protection of the fuel building is achieved by full hardening. The inner building structures are decoupled from the outer protection wall in order to ensure the integrity of the spent fuel pool. The new fuel assemblies are stored in the fuel building to enable easy access thereto. Inside the reactor building, a loading machine transfers the spent as well as the new fuel assemblies into or out of the reactor. A transfer station enables the transfer of the fuel elements from the fuel building to the reactor building and vice versa.

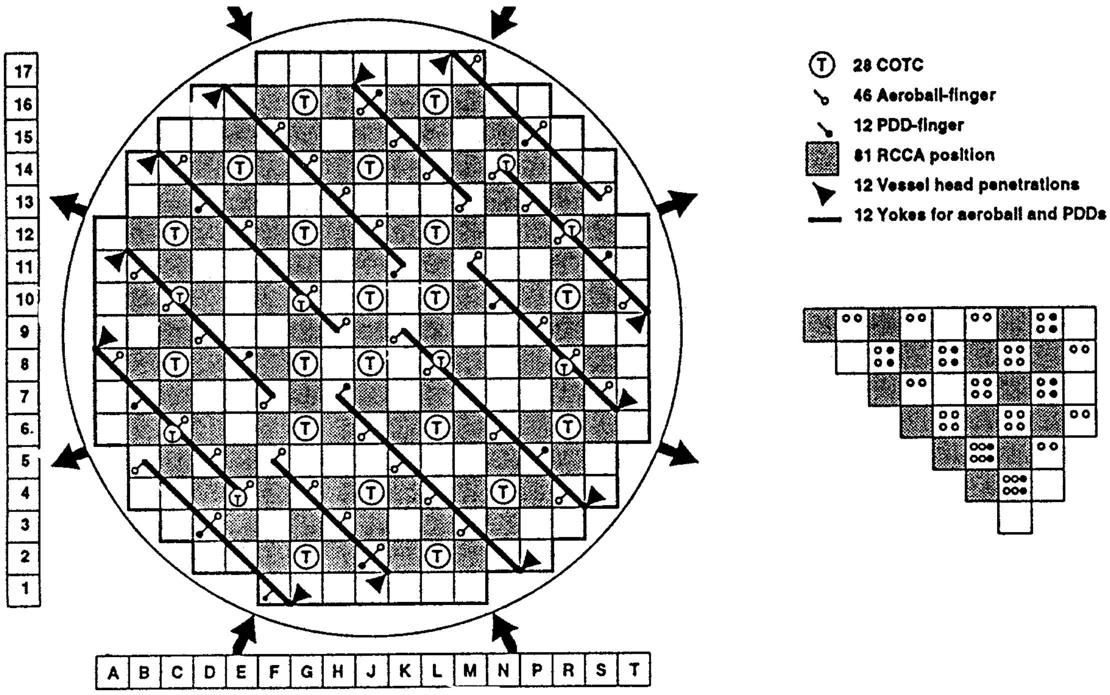

FIG. 4.6-1. EPR - Core instrumentation.

4.6.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel (RPV) is designed for a lifetime 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. There is a high safety margin against brittle fracture in the design of the RPV based on the material and the low total neutron fluence. The upper part of the RPV will be manufactured 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 RPV is internally clad with two layers of low carbon stainless steel. 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. Inspection from the outside will be performed.

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. As material for the heating tubes Incoloy 800 has been chosen for the further studies. The possibility to use Inconel 690 is kept open. Both materials have proven excellent properties regarding corrosion resistance and are exchangeable without affecting steam generator design parameters. The heating tubes are supported by perforated plates. With respect to the pressure boundary, the same material as for the reactor pressure vessel will be chosen.

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. In addition, in order to accommodate a lifetime of 60 years and meet the plant power flexibility requirements, several separate spray lines are provided and operated alternatively. 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 stainless steel and welded with an Inconel material.

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.

Main coolant lines

The main coolant lines will be forged stainless steel. Additionally the break preclusion concept will be applied. This concept consists of a high quality in design, construction and surveillance and enables to rule out a catastrophic failure of a main coolant line as regards its possible mechanical effects. Consequently pipe whip restraints are not more necessary. 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 and of the containment design.

Reactor internals

The core barrel flange rests on a ledge machined from the flange of the reactor pressure vessel and is preloaded axially by an elastic system. The fuel assemblies are placed directly on a flat perforated plate, machined from a forging of stainless steel and welded all around to the core barrel. The fuel assemblies are centered by two pins each, which are screwed to the core plate. 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 coolant 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 to the reactor pressure vessel wall and to flatten the power distribution in the core. This structure is called the heavy reflector and aims at savings concerning the enrichment requirements. 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.

4.6.2.5 Reactor auxiliary systems

Chemical and volume control system

The chemical and volume control system (CVCS) is mainly an operational system with the tasks to control the water inventory, the water quality and the boron concentration in the primary system. Additionally, the system adjusts the chemical composition of the reactor coolant system and removes dissolved gases by degasification of the letdown flow. In addition, this system provides a means for pressure control of the primary circuit by supplying auxiliary spray to the pressurizer, thereby condensing steam and reducing primary system pressure, and seal water to the reactor coolant pumps.

In the EPR, this system has also one safety-related task in the frame of risk reduction, which is to ensure capability of auxiliary spray of the pressurizer as it essential to avoid core uncovery in case of the total loss of feedwater.

Component cooling water system

The component cooling water system (CCWS) ensures primarily the following safety functions:

- Heat removal from the safety injection/residual heat removal system to the essential service water system (ESWS);
- Heat removal from the fuel pool cooling system to the ESWS;
- Cooling of the thermal barrier of the reactor coolant pumps;
- Heat removal from safety classified chillers; and
- Cooling of the CVCS (in the frame of risk reduction).

The functions of the CCWS are achieved by use of safety classified heat exchangers and adequate components such as pumps and valves.

Essential service water system

The essential service water system (ESWS) includes the following safety functions:

- Cooling of the CCWS; and
- Cooling of the fuel pool cooling system as long as any fuel assemblies are in the spent fuel storage pool outside the containment.

The ESWS consists of four heat exchangers that in particular have to cool components of safety systems.

4.6.2.6 Operating characteristics

The EPR is a nuclear island for an electrical power *output* of about 1550 MW(e). The primary components are enlarged relative to current types, the safety and operating systems have been redesigned or updated to the conditions required.

The EPR is designed for operation between 20 and 100% of rated generator power. In the power range between 60% 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. In the lower power range, below 60% power, the main steam pressure is kept constant 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, and in minimizing the loads on the pressurizer surge line and the control rod drive mechanisms during load changes in the most frequent operation mode in the upper power range. In the lower power range, a continuation of the constant average coolant temperature mode would penalise the secondary side design (by a design pressure increase) and eventually also the primary side design pressure. Therefore the constant main steam pressure operating mode is implemented for this lower power range.

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 follow capability can briefly be summarised as:

- ± 5%/min ramp load change within 50 and 100% of rated thermal power (±2.5%/min within the 25 to 50 % range);

- ±10% step load change within 20 and 100% of rated power;
- + 20% power increase within 2 minutes;
- 100-25-100% load follow operation cycles, with several load changes per day; and
- Primary and secondary grid frequency control equivalent to ~ ±10%.

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.

4.6.3 Description of turbine generator plant system

The EPR project is basically restricted to the nuclear steam supply system (NSSS) design, intended to be a common design for France and Germany.

The strategy adopted for the turbine generator and its associated systems is to improve plant efficiency by utilization of high-performance components; the extra costs associated with high-performance components would be offset by the gain in efficiency.

4.6.3.1 Turbine generator plant

The turbine of the EPR is a development based on the Arabelle turbine of the N4 or on the Konvoi turbine. The net power output is some 1550 MW(e). The saturated steam coming from the steam generators is flowing to the inlet valves of the high pressure (HP) turbine via four steam lines. After expansion in the HP turbine, the steam is routed through two steam moisture separator and reheater units to the three low pressure turbines in parallel. After expansion in the low pressure turbines, the steam goes to the condenser, which is of modular type for easy maintenance, with two modules for each low pressure cylinder. Depending on the site characteristics, the tube material will be stainless steel for river sites or titanium for coastal sites.

The condenser cooling water system is equipped with two or three motor-driven pumps capable of supplying 100% of the nominal cooling flow. In case of pump malfunction, the turbine is not tripped, and the power output is decreased to match the remaining flow.

The turbine bypass to the condenser is designed to accept 50% of the rated steam flow to the turbine.

The generator is a four-pole type, using hydrogen as the rotor coolant and water for cooling of the stator windings. The rotor is directly coupled to the turbine.

4.6.3.2 Condensate and feedwater systems

Water coming from the condenser is pumped through four low pressure heaters to the deaerator and the storage tank by means of two or three (one of them in standby modus) condensate pumps supplying 100%. Three high pressure electric motor-driven pumps, each with a capability of providing 50% of the rated flow (one pump is in standby) circulate the feedwater through the high pressure heaters for injection in the steam generators.

4.6.4 Instrumentation and control systems

4.6.4.1 I&C design concept, including control rooms

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 (short I&C) are divided up into classes 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.

During the basic design stage, it has been ascertained that different I&C functions are assigned to the proper class, and that any I&C function with a higher classification has priority over lower classified (less important) ones. Failures in a lower classified function are not allowed to jeopardize functions of a higher classification.

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.

Safety I&C

The safety I&C systems and equipment shall have a high reliability so that they will not be a dominant contributor to the unavailability of safety functions. 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 are installed in divisions with physical separation and with a minimum number of interconnections. Interconnections are energetically isolated against over-voltages from a disturbed division or train (e.g. by means of fibre optics), and erroneous signals from a disturbed train are prevented from affecting the other trains by means of majority voting. Necessary safety actions must be performed from the undisturbed trains independent of the state of a disturbed train or division.

Appropriate measures are provided to cope with common cause failures (CCF) in order to meet the overall probabilistic design targets. Special probability values for digital I&C are practically nonexistent, 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 independent from the operational I&C when interconnections cannot 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. Signal exchange between safety I&C systems and operational I&C systems is energetically isolated.

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.

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 backup 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 three 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 backup 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 backup means. The safety control area (with the backup 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. 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.

4.6.5 Electrical systems

The main features of the electrical systems are:

- Power delivery to 400 kV main grid via one (variant A) or two (variant B) generator breaker(s) and main transformer(s);
- Two auxiliary transformers for power supply from the main generator or the main grid to the auxiliary loads;
- One standby transformer for power supply from the standby grid to auxiliary loads;
- Four train arrangement for the operational and emergency power supply to Conventional and Nuclear Island;
- Four train, four division concept for the emergency power supply in accordance with fluid system design;

- Four diesel generator sets (10 kV) in separated buildings;
- Two additional "small" diesel generator sets (690 V) for Station Blackout mitigation;
- AC voltage levels 10 kV, 690 V and 400 V;
- Two battery sets (220 V DC) in Conventional Island;
- Four battery sets (220 V DC) in Nuclear Island; and
- I&C power supply via AC/DC converters.

Emergency power supply

The emergency power supply is arranged in a four train, four division concept. The safety loads and some non-safety loads are connected to the emergency power supply system. The safety loads correspond to the items, which are necessary to shut down the reactor safely, keep it in shutdown condition, remove the residual heat and stored heat and to prevent impermissible release of radioactive substances, under accidental conditions. A direct connection between the emergency and normal power supply allows a simple and safe separation from the normal supply in case of emergency power mode. The emergency power supply system is equipped with four separate and independent diesel generator units (Emergency Diesel Generators), which are used in case of emergency power mode. The startup time of Emergency Diesel Generators is in accordance with the requirements of the supplied processes. The autonomy requirement is equivalent to three days at full power.

Station blackout power supply

The station blackout (SBO) power supply is part of the emergency power supply system and arranged in a two train, two division concept. In case of loss of both the off-site power supply and emergency diesel generators, the loads, which are necessary to safe shutdown of the plant, are connected to the SBO power supply. The Station Blackout power supply system is equipped with two separate and independent diesel generator units (SBO diesel generators), which are diversified in regards to the emergency diesel generator sets. During the first 2 hours of SBO the batteries take over the supply of the needed consumer, before the end of the battery discharge time of 2 hours the SBO diesel generators have to be started. The startup of station blackout generators is manual from the Main Control Room. The autonomy requirement is equivalent to 24 hours at full power.

Uninterrupted power supply and distribution system

The uninterrupted power supply is part of the emergency power supply system and arranged in a four train, four division concept. The uninterrupted power supply system provides continuous and reliable electrical power for I&C loads such as the plant protection systems and to other loads The batteries constitute the on-site power source for operation of the loads of the low-voltage uninterrupted AC systems. The rectifiers have sufficient capacity to restore the batteries from a discharged condition to a minimum charged state within an acceptable time while at the same time supplying the largest combined demands of the various steady state loads following the loss of normal power.The battery autonomy requirement is equivalent to 2 hours at full load.

4.6.6 Safety concept

4.6.6.1 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.

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 envelope anticipated operational occurrences, infrequent and limiting accidents.

Stringent radiological limits are applied for normal operation and anticipated operational occurrences as well as for accidents.

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 relates 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 into the containment, 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).

RRC-A is introduced in order to define a limited number of additional design means necessary to meet the overall probabilistic targets.

The assessment of RRC-B features for considerable time will be deterministic, supported by level 1+ and level 2 PSA.

Probabilistic targets

The overall safety objective that has been set for the EPR, requires that the probability of core melt frequency (CMF) shall be below 10^{-5}/year including all events and all reactor operating states.

In order to meet this objective, some specific probabilistic design targets have been defined for the design phases:

- For internal events, the CMF shall be $< 10^{-6}$ per year; respectively for the power states and for the shutdown states; and
- In the course of the design process it is convenient to use a CMF target per family of initiating events a design target of 10^{-7} per year and per family.

External and internal hazards

External and internal hazards are normally not assigned directly to specific plant condition categories or risk reduction categories, in order to avoid the study of numerous sequences. But the main principles behind the deterministic design basis and the risk reduction approach (namely: the more probable the event, the more conservative rules and acceptance criteria) are applied also for dealing with external and internal hazards.

The following internal hazards are covered by the EPR design:

- Pipe leaks and breaks, failure of vessels, tanks, pumps and valves;
- Flooding;
- Internal missiles;
- Drop of loads;
- Internal explosions; and
- Fire.

The main external hazards considered in the EPR design are:

- Earthquake;
- Airplane crash; and
- Explosion pressure wave.

External hazards are to a certain extent site-dependent. The possibility of choosing the boundary conditions in such a way, that it should be possible to construct the EPR on most potential sites is being considered. Sites with an extreme external hazard potential are not taken into consideration as potential sites.

4.6.6.2 Safety systems and features (active and passive)

Safety systems configuration

Important safety systems (safety injection, emergency feedwater, main steam relief, cooling chain, emergency electric power) are arranged in a four train configuration as depicted in Figure 4.6-2.

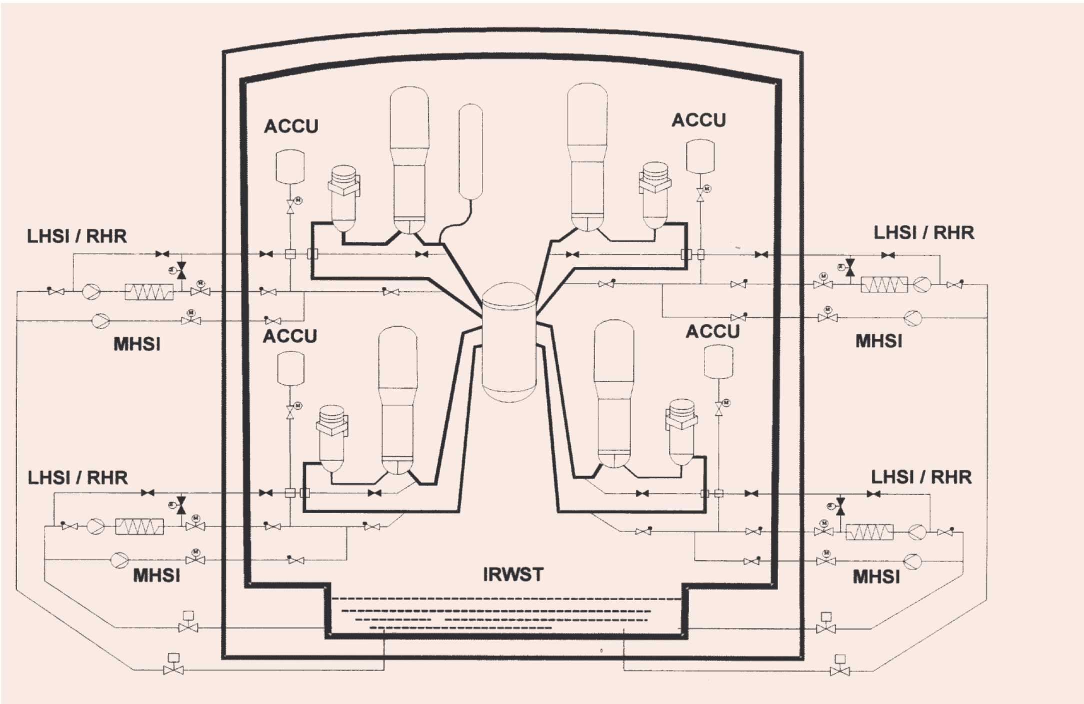

FIG. 4.6-2. EPR - Configuration of fluid safety systems.

The layout comprises four separate divisions, corresponding to the four trains. A simple and straightforward 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.

The safety injection systems, for which an overview is presented in Table 4.6-I, feature an in-containment refuelling water storage tank (IRWST) located at the bottom of the containment and provide injection in both hot and cold legs of the RCS. During design basis accidents the low-head safety injection (LHSI) system transfers the decay heat to the ultimate heat sink via heat exchangers. The containment spray system foreseen is only used for heat removal in case of severe accidents. 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.

TABLE 4.6-I. ORGANIZATION OF SAFETY INJECTION 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

TABLE 4.6-II. DIVERSE SYSTEM JUNCTIONS FOR MAJOR 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		

The function of any one of the safety systems can be accomplished by another diverse system (or group of systems) in the event malfunctions, as shown in Table 4.6-II. In the EPR design efforts have been devoted to prevent high pressure core melt scenarios. Prevention of such scenarios implies need of a highly reliable secondary side heat removal system.

Detailed investigations of active versus passive systems have led to 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.

Safety injection systems / Residual heat removal system

The residual heat removal (RHR) system is combined with the low head safety injection system (LHSI) of the safety injection system (SIS). 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 MHSI system is 8.5 MPa. This shut-off head is sufficient to cope with all LOCA related requirements, since a reliable secondary side partial cooldown is provided via safety-grade main steam relief valves.

In addition to the MHSI system, cold leg accumulator injection is provided to cope with large and intermediate break sizes. Four accumulators with a volume of 50 m^3 each are provided, each directly assigned to one cold leg. The response pressure of the accumulators is designed to 4.5 MPa.

The low head safety injection (LHSI) 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 or pure hot leg injection 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.

In addition to these accident mitigation functions, the low pressure injection system is designed to be part of the operational residual heat removal system at low RCS temperatures.

In-containment refuelling -water storage tank

The in-containment refuelling water storage tank (IRWST) 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 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 IRWST. In the case of severe accidents the IRWST 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.

Emergency feedwater system

The emergency feedwater (EFW) system consists of four separate and independent trains, each providing injection to one of the four steam generators. Each emergency feedwater pump takes suction from an emergency feedwater pool. These pools 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 maximum extent.

For startup and shutdown an additional dedicated system is installed, called start-up and shutdown system (SSS). The SSS 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 small 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 pool 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 at the system after sufficient grace period.

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.

Residual heat removal system

The residual heat removal (RHR) system is designed to transfer residual heat from the reactor coolant system via the cooling chain consisting of the component cooling water system and essential service water system to the 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 RHR system consists of four independent trains each of which uses the LHSI pump and LHSI heat exchanger in the discharge path of the coolant towards the reactor coolant system (RCS). The pump discharge is routed via LHSI heat exchangers to a cold leg of the RCS. A bypass line of the heat exchanger is provided to allow control of the cooldown rate. The LHSI system heat exchangers are cooled by the component cooling water (CCW) system train, which is located in the same division as the associated residual heat removal train. Switch-over from secondary side cooling to RHR cooling is foreseen at an average reactor coolant system temperature ≤ 120°C. For cooldown from 120°C to 100°C only the SIS trains 1 and 4 should be used, train 2 and 3 will be used additionally only below 100°C. During normal operation, only two RHR trains are used for cooldown and cold shutdown. All four trains are not used unless the RCS temperature is below 100°C.

In case of a break in one of the RHR trains, the affected train is automatically isolated.

Extra boration system

The safety function of the extra boration system (EBS) is to ensure for any PCC or RRC-A event the capability for a sufficient boration of the RCS to allow the transfer to the safe shutdown state.

The EBS consists of two separate and independent trains, each providing capability of injection of the total amount of concentrated boric acid, at 7000 ppm required for reaching cold shutdown from any steady state power operation.

One of the two EBS trains can be used for the periodic hydrostatic test of the RCS.

4.6.6.3 Severe accidents (beyond design basis accidents)

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 licensing authorities tend to take risk reductions in the event of more and more accidents into account, in particular with respect to possible need for relocation or evacuation of the population beyond the immediate vicinity of the plant, and restrictions on the use of foodstuff beyond the first year harvest.

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 utterly important, and this 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 base mat melt-through, ultimate pressure resistance to cope with energetic events;
- Pressure reduction inside the containment by dedicated heat removal; and
- Collection of unavoidable containment leakages in the annulus atmosphere and release via the stack after filtration.

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 with the potential consequence of early containment failure. 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.
- Prevention respectively reduction of the hydrogen-concentration in the containment by catalytic H2-recombiners. 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 and a special cooling device (Figure 4.6-3).
- 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 (covered with sacrificial material), which will resist melt-through for a certain time, in order to accumulate the melt and to achieve the necessary melt conditioning in the pit.
- Provisions for connecting the spreading compartment with the in-containment refuelling water storage tank (IRWST) for flooding the melt including the cooling device after spreading; these pipe connections are closed during normal operation.
- The special cooling device supplied with water, either passively from the in-containment refueling water storage tank or actively from the containment heat removal system to ensure integrity of the protective layer, and to prevent high temperature loadings in the base mat.

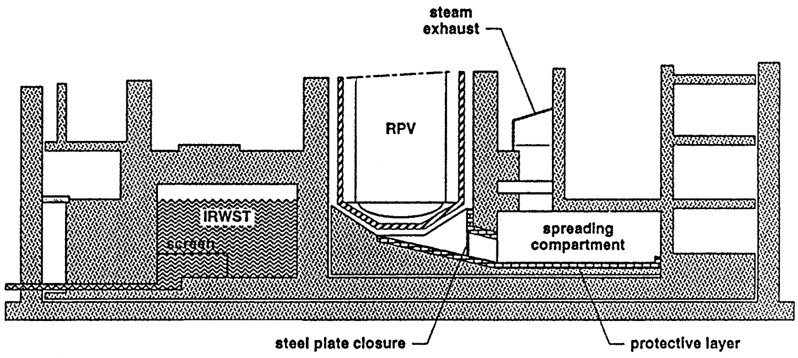

FIG. 4.6-3. EPR - Retention of molten corium.

4.6.7 Plant layout

The plant layout is governed by a number of principles derived from the 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.

4.6.7.1 Buildings and structures, including plot plan

The general layout of the EPR plant is shown on Figure 4.6-4. The reactor building with the containment is surrounded by the safeguard and fuel buildings which contain the safety systems. Most of the safety-grade systems are designed with a four-fold redundancy, arranged in four independent divisions with complete physical separation. Each division comprises a low head injection 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.

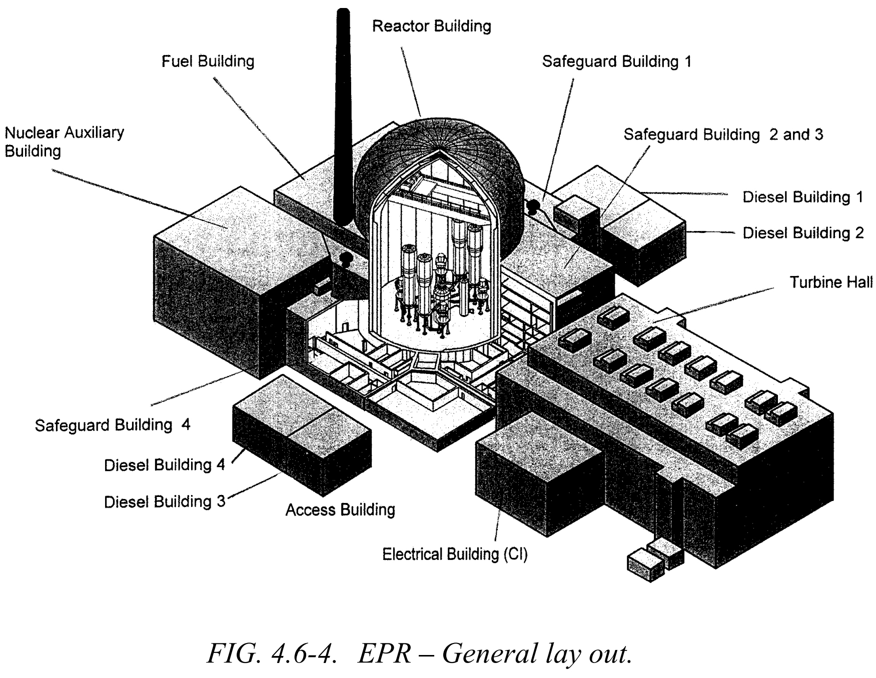

FIG. 4.6-4. EPR – General lay out.

Design requirements

The plant is designed to withstand the impacts of internal and external events, as specified below. With respect to earthquake and explosion pressure waves, the buildings and structures have been strengthened so that collapsing structures will not jeopardize the function of safety-grade equipment, 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.

Internal hazards: The loads from internal events (e.g. fire loads, missile loads, jet impingement loads, flooding effects) are included in the design. For the overall plant layout they have been minimized and easy protection measures have been chosen to cover sensitive equipment.

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.

Besides the requirements concerning severe accident mitigation, the application of radiation protection principles influences the plant layout significantly. The EPR is designed for a separation of hot (controlled) and cold (not controlled) areas.

The reactor building and the fuel building are classified as hot zones. Within the safeguard buildings, the safety injection system part is arranged in the inner areas, which are classified as hot zones, whereas the component cooling and emergency feedwater systems are installed in the outer areas, which are classified as cold zones.

4.6 7.2 Reactor building

The reactor building (Figures 4.6-5 and 4.6-6) 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.

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.

4.6.7.3 Containment

Adoption of a double concrete containment design was decided for the EPR. The particular design concept uses, for the inner containment wall, the prestressed concrete technology. The leak-tightness requirement for the inner containment wall is less than 1 % volume per day. The outer wall, in reinforced concrete with a thickness of 1,30 m, completes the double containment arrangement.

To ensure overall 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 with double sealings on both sides.

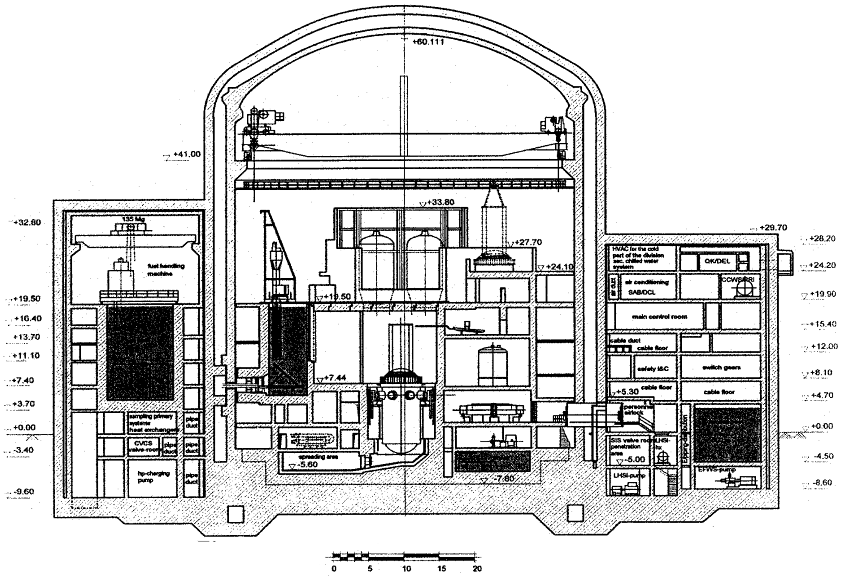

FIG. 4.6-5. EPR - Building arrangement, section A-A.

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 (removing heat).

Accumulation of combustible gases, especially hydrogen, is controlled. Furthermore, protective layers protect the base mat in the spreading compartment and the dedicated cooling system fed by the containment 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 conditions 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 pre-stressed concrete inner wall will also ensure capability to perform an integral leakage pressure test in air at design pressure, thus providing positive proof of containment structural and leak-tightness capability for the entire range of pressure of all severe accident scenarios.

Containment 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.

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, dedicated melting plugs that allow water to cool the molten core material by passive means provide water ingress. The generated steam is condensed by a containment heat removal system exclusively provided for these accident sequences.

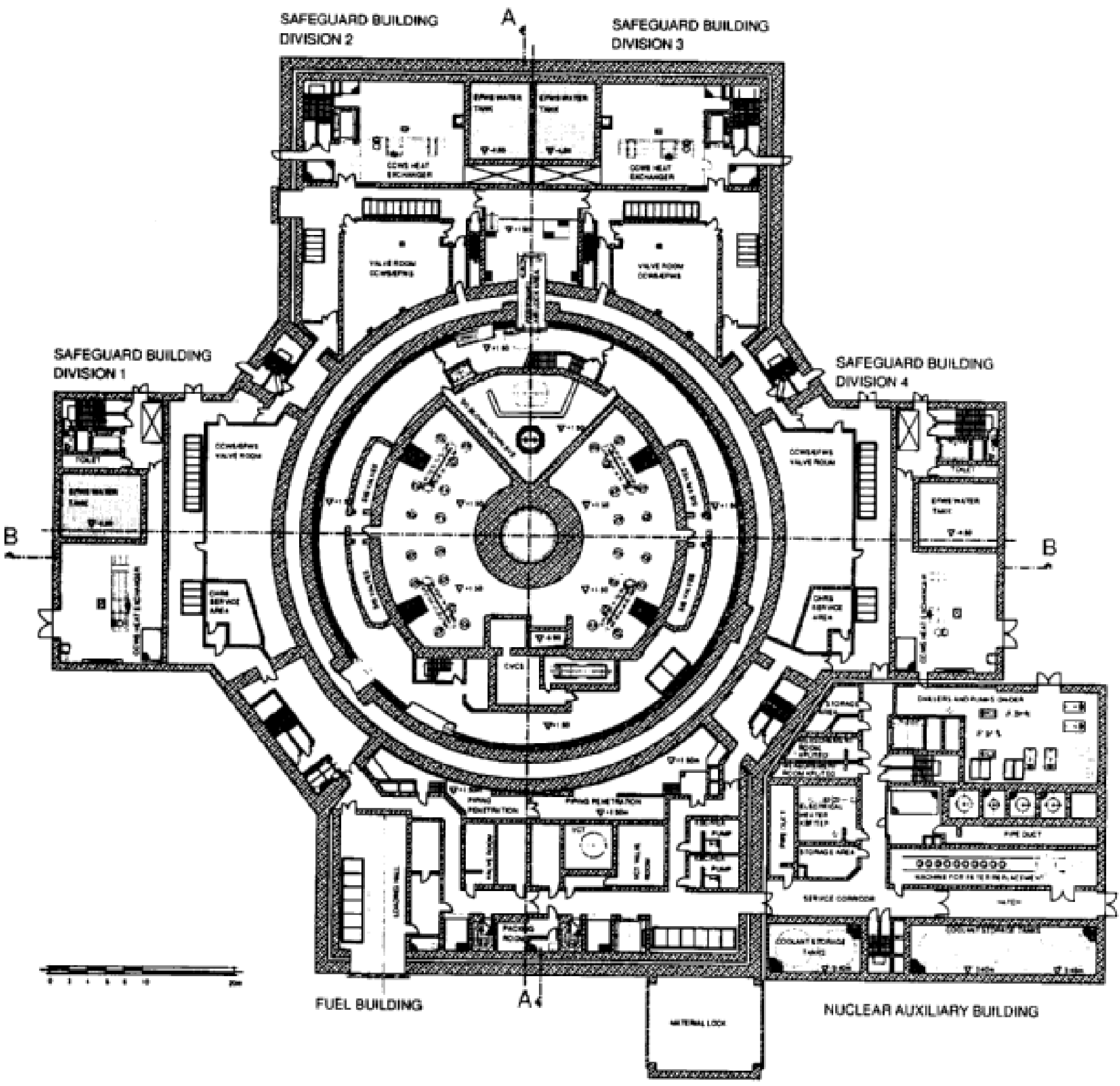

FIG. 4.6-6. EPR - Building arrangement, plan view ± 0,00.

A high temperature resistant protective layer on the reactor pit floor and the spreading area prevents interaction between concrete and the molten core material.

4.6.8 Technical data

General plant data		
Power plant output, gross	1650	MWe
Power plant output, net	~ 1 550	MWe
Reactor thermal output	4 250	MWt
Power plant efficiency, net	~ 36	%
Cooling water temperature		°C
Nuclear steam supply system		
Number of coolant loops	4	
Primary circuit volume, including pressuriser	380 + 75	m^3
Steam flow rate at nominal conditions		kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure		°C/MPa
Feedwater temperature/pressure		°C/MPa
Reactor coolant system		
Primary coolant flow rate BE	23150	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	295,5	°C
Coolant outlet temperature, at RPV outlet	328,1	°C
Mean temperature rise across core		°C
Reactor core		
Active core height	4.2	m
Equivalent core diameter	3767	mm
Heat transfer surface in the core	7975	m^2
Fuel inventory		tU
Average linear heat rate	14.95	kW/m
Average fuel power density	5	kW/kg U
Average core power density (volumetric)	89,3	kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		

Fuel material	Sintered	UO_2/MOX
Fuel assembly total length	4800	mm
Rod array	Square, 17x17	
Number of fuel assemblies	241	
Number of fuel rods/assembly	265	
Number of control rod guide tubes/assembly	24	
Number of spacers	10	
Enrichment (range) of first core	≤ 5.0	Wt%
Enrichment of reload fuel at equilibrium core		
Operating cycle length (fuel cycle length)	18	Months
Average discharge burnup of fuel	60 000	MWd/t
Cladding tube material	Zircaloy	
Cladding tube wall thickness	0,57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly		kg
Active length of fuel rods	4200	mm
Burnable absorber, strategy/material	Gd_2O_3	
Number of control rods	89	
Absorber rods per control assembly	24	
Absorber material: Upper part	Ag-80, In-15,Cd-5	
Lower part	B_4C	
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	
Total height	12700	mm
Base material: cylindrical shell	MnMoNi steel	
RPV head		
liner		
Design pressure/temperature	17.6/351	Mpa/ °C
Transport weight (lower part		t
RPV head		t

Steam generators

Type	U-tube heat exchanger	
Number	4	
Heat transfer surface	7960	m^2
Number of heat exchanger tubes	5980	
Tube dimensions outer diameter	19,05	mm
Maximum outer diameter		mm
Total height	24164	mm
Transport weight		t
Shell and tube sheet material	16MND5 (20MnMoNi55)	
Tube material	Incoloy 800 (Inconel 690)	

Reactor coolant pump

Type	Single-stage, centrifugal pump	
Number	4	
Design pressure/temperature		Mpa/°C
Design flow rate (at operating conditions)	5 475	kg/s
Pump head		
Power demand at coupling, cold/hot		
Pump casing material	Stainless or Ferritic with cladding 3 seals	
Pump speed		rpm

Pressuriser

Total volume	~ 75	m^3
Steam volume: full power/zero power		m^3
Design pressure/temperature	17.6/362	MPa /°C
Heating power of the heater rods	2592	kW
Number of heater rods	108	
Inner diameter	2820	mm
Total height	13915	mm
Material	16MND5 (20MnMoNi55)	
Transport weight		t

Pressuriser relief tank (if any)

Total volume		m^3
Design pressure/temperature		Mpa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Prestressed concrete	
Overall form (cyl.)	cylindrical	
Dimensions (diameter/height)	46,8/	m
Free volume	80000	m^3
Design pressure/temperature (DBEs)	~ 650/	kPa/°C
(severe accident situations)	~ 650/	kPa/°C
Design leakage rate	< 1	vol%/day
Is secondary containment provided?	Yes, reinforced concrete (APC-Protection)	

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage		kV
	rated capacity		MVA
Plant transformers,	rated voltage		kV
	rated capacity		MVA
Start-up transformer	rated voltage		kV
	rated capacity		MVA
Medium voltage busbars (6kV or 10 kV)			
Number of low voltage busbar systems			
Standby diesel generating units: number			
	rated power		MW

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

<u>Turbine plant</u>

Number of turbines per reactor	1	
Type of turbine(s)		
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HMP/ 3LP / 1 HP/3LP	
Turbine speed	1500	rpm
Overall length of turbine unit	68.77 / ~70	m
Overall width of turbine unit	18 / 15,5	m
HP inlet pressure/temperature		MPa /°C

<u>Generator</u>

Type	Two-pole, synchronous	
Rated power	1900 / 1950	MVA
Active power	1600 / 1755	MW
Voltage	20 / 26	kV
Frequency	50	Hz
Total generator mass	> 500 / 565	t
Overall length of generator incl. exciter	~23,5	m

<u>Condenser</u>

Type [for once-through operation]	Modular /	
Number of tubes	84 400 /	
Heat transfer area	86 763 /	m^2
Cooling water flow rate	43 /	m^3/s
Cooling water temperature	13 /	°C
Condenser pressure	51.5 / 45	hPa

<u>Condensate pumps</u>

Number	2x50% / 3x50%	
Flow rate	670 / 2 x ~685	kg/s
Pump head	365	mWG
Temperature	40 / 30	°C
Pump speed		rpm

<u>Condensate clean-up system</u>

Full flow/part flow		
Filter type		

<u>Feedwater tank</u>

Volume	830 / <400	m^3
Pressure/temperature	~10/180	MPa/°C

<u>Feedwater pumps</u>

Number	3x50% / 4x35%	
Flow rate	2x1200 / 3x980	kg/s
Pump head	850	m WG
Feedwater temperature	230	°C
Pump speed [variable, full speed =]		rpm

<u>Condensate and feedwater heaters</u>

Number of heating stages	6 (4 LP + 2 HP / (5+2)	7
Redundancies		

4.6.9 Measures to enhance economy and maintainability

- Elimination of common mode failures by physical separation and diverse backup functions for safety functions;
- Increase of grace periods for operator actions by designing components (e.g. pressurizer and steam generators) with larger water inventories to smoothen transients;
- Less sensitivity to human errors by an optimized man-machine interface by digital instrumentation and control systems and status-oriented information supplied by modern operator information systems;
- High availability. Lifetime of the plant 60 years; and
- Preventive maintenance features are incorporated in the design from the beginning to minimize outage durations.

4.6.10 Project status and planned schedule

The joint French and French programme has already been running some years. In the recent years, the work was mainly concentrated on the conceptual design and the overall safety strategy. Since 1993, the concept proposals are subjected to assessment of the authorities of both countries. The French and German Safety Authorities and safety experts worked closely together in order to ensure further the outstanding safety standard in France and Germany. Their major objective was the establishment of a set of common rules and regulations in order to converge the French and German licensing requirements. During the Basic Design, they actively and intensively reviewed the EPR safety concept on the basis of their jointly issued "Proposal for a common safety approach for future pressurized water reactors". This work was concluded in October 2000 by the validation of the Technical Guidelines for the design and construction of future Nuclear Power Plants with PWRs.

Several major milestones have been achieved:

1) The safety authorities of both countries have positively accomplished this assessment of the main safety issues.
2) EdF and the German utilities have, on 23 February 1995, awarded a contract to NPI and its parent companies concerning the Basic Design Engineering. The Basic Design of the EPR was completed successfully end of 1999.
3) The design was evaluated against the European utility requirements (EUR) and approved by the EUR steering committee.
4) The target of the present work is mainly the preparation of the Preliminary Safety Analysis Report and the deepening of some Basic Design subjects related to the NSSS.
5) The EPR is being offered as the PWR plant of Framatome ANP to the Finish utilities in fall 2002

Bibliography

Comprehensive presentations of the EPR plant design and project took place at the Palais de la Musique et de Congres, Strasbourg, 13–14 November 1995, and in the Hotel Maritim, Cologne, 19–21 October 1997, both organized jointly by Societe Francaise d'Energie Nucleaire, France (SFEN) and Kerntechnische Gesellschaft, Germany (KTG).

The lectures given are compiled in the proceedings entitled “The EPR Project”, Strasbourg, Palais de la Musique et de Congres, 13–14 November 1995, and « The European Pressurized Water Reactor EPR », 19–21 October 1997.

4.7 ESBWR (GENERAL ELECTRIC, USA)

4.7.1 Introduction

General Electric's ESBWR is a 1390 MW(e) power plant design based on the earlier 670 MW(e) SBWR design. The ESBWR reactor core has a rated thermal output of 4000 MW(th). Like the earlier SBWR design, the ESBWR design incorporates innovative, yet proven, features to further simplify an inherently simple direct cycle nuclear plant.

The ESBWR design objectives include:

- a 60 year plant life from the date of full power operating license;
- 92% or greater plant availability;
- 12–24 month refueling intervals;
- a personnel radiation exposure limit of 50 manrem/year;
- providing safety related functions primarily through passive means;
- a core damage frequency of less than 10^{-6} per reactor year;
- limiting significant release frequency, from all events (internal & external), to 5 x 10^{-8} per reactor year;
- requiring no operator action on safety systems, for 72 hours following a design basis accident, to maintain the reactor and containment at safe stable conditions.

The principal design criteria governing the ESBWR plant design are associated with either a power generation function or a safety related function. Each is discussed below.

General power generation design criteria

The plant is designed to produce electricity from a turbine generator unit using steam generated in the reactor. Heat removal systems are provided with sufficient capacity and operational capability to remove heat generated by the reactor core for the full range of normal operating conditions and off-normal transients. Backup heat removal systems are provided to remove reactor core decay heat in circumstances where the normal heat removal systems have become inoperative. The capacity of the backup systems is adequate to prevent fuel cladding damage.

In conjunction with other plant systems, the fuel cladding is designed to retain its integrity for the design life of the fuel. The consequences of plant system failures (such as pipe break, etc.) therefore remain within acceptable limits throughout the range of normal operating conditions and off-normal transients.

Control equipment is provided to allow the reactor to respond automatically to normal load changes and off-normal transients. Reactor power level is manually controllable, with interlocks or other automatic equipment provided as backup to procedural control to avoid conditions requiring the functioning of safety related systems or engineered safety features.

General safety design criteria

The plant is designed, fabricated, erected and operated in such a way that the release of radioactive material to the environment will not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operation, for off-normal transients and for accidents.

The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient. The reactor is designed with abundant core coolant flow so that there is high flow margin to prevent divergent oscillation of any operating characteristics considering the interaction of the reactor with other appropriate plant systems.

Safety related systems and engineered safety features function to ensure that no damage occurs to the reactor coolant pressure boundary from internal pressures resulting from off-normal transient or accident conditions. Where positive, precise action is immediately required in response to off-normal transients or accidents, such action is automatic and requires no decision or manipulation of controls by plant operations personnel. The design of safety related systems, components and structures includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the plant site.

Standby batteries are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available. The standby batteries have sufficient capacity to concurrently power all safety related systems requiring electrical power.

The ESBWR design has a pressure suppression-type containment that completely encloses the reactor system, drywell, suppression chamber, and certain other associated volumes. This containment, in conjunction with the reactor building and other safety related features, limits the radiological effects from design basis accidents to less than the prescribed regulatory limits. A perspective of the containment and its contents is shown in Figure 4.7-1.

To maintain the integrity of the containment system, provisions are made for removing energy released to it under accident conditions. In the event of a design basis loss of coolant accident (LOCA), emergency core cooling is provided to keep the core covered with coolant and to limit fuel cladding temperatures to far less than the regulatory limit of 1200°C. The emergency core cooling system provides core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary piping. When required, emergency core cooling is initiated automatically, regardless of the availability of power from the normal plant generating system or offsite supplies.

The control room is shielded against radiation so that continued occupancy under design basis accident conditions is possible. In the event that the control room becomes uninhabitable for other reasons, it is possible to bring the reactor from power range operation to cold shutdown conditions by utilizing alternative controls and equipment that are available outside the control room.

Fuel handling and storage facilities are designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel as necessary to meet operating and offsite dose constraints

4.7.2 Description of the nuclear systems

4.7.2.1 Primary circuit and its main characteristics

The primary functions of the nuclear boiler system are:

- to deliver steam from the reactor pressure vessel (RPV) to the turbine main steam system;
- to deliver feedwater from the condensate and feedwater system to the RPV;
- to provide overpressure protection of the reactor coolant pressure boundary (RCPB);
- to provide automatic depressurization of the RPV in the event of a loss of coolant accident (LOCA) where the RPV does not otherwise depressurize rapidly;
- to provide (with the exception of neutron flux monitoring) the instrumentation necessary for monitoring RPV conditions such as pressure, metal temperature, and water level .

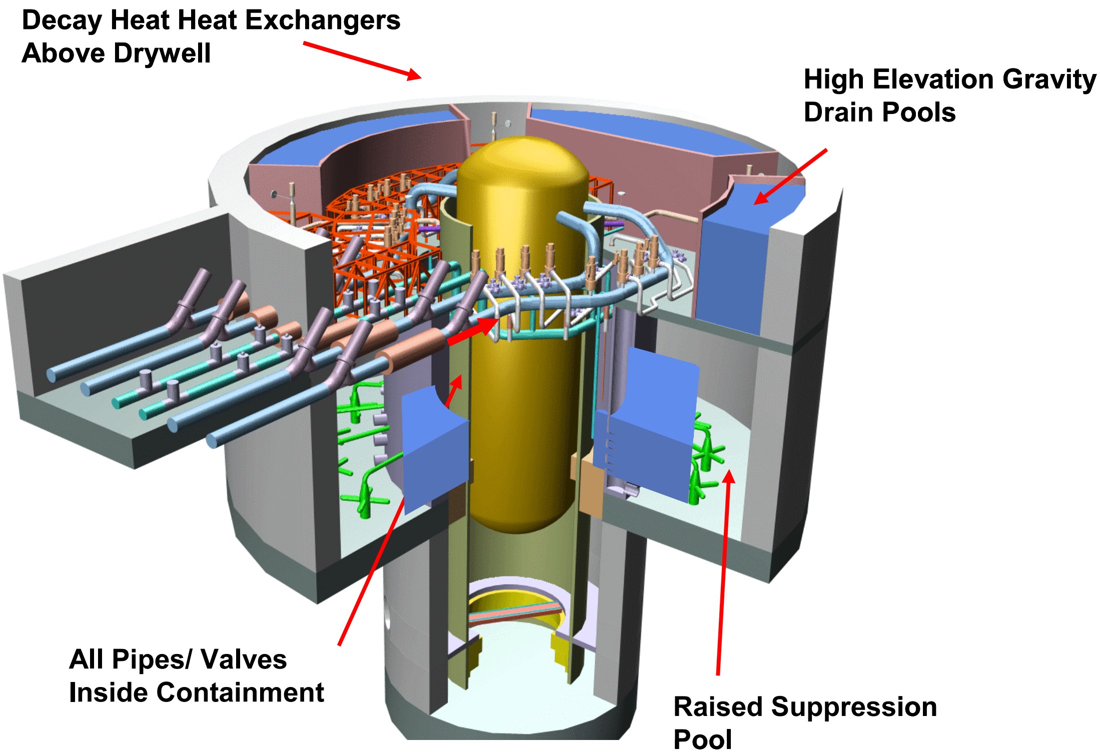

FIG. 4.7-1. ESBWR containment envelope and design features.

The main steam lines (MSLs) are designed to direct steam from the RPV to the turbine, and the feedwater lines direct feedwater from the condensate and feedwater system to the RPV.

The main steam line flow limiter, a flow restricting venturi built into the RPV MSL nozzle of each of the four main steam lines, limits the coolant blowdown rate from the reactor vessel to a (choked) flow rate equal to or less than 200% of rated steam flow in the event a main steam line break occurs anywhere downstream of the nozzle.

There are two main steam isolation valves (MSIVs) welded into each of the four MSLs, one inner MSIV in the containment and one outer MSIV outside the containment. The MSIVs are Y-pattern globe valves. The Y-pattern configuration permits the inlet and outlet flow passages to be streamlined to minimize pressure drop during normal steam flow and allows reactor pressure to assist in valve closure and seating.

Overpressure protection of the RPV is provided by safety/relief valves (SRVs) located on the four main steam lines (MSLs), between the RPV and the inboard MSIV. Discharge lines from each of three SRVs per MSL are routed to quenchers in the suppression pool. The SRVs also provide a depressurization capability as part of the automatic depressurization subsystem (ADS).

The ADS consists of the 12 above described SRVs, plus 8 depressurization valves (DPVs) and their associated instrumentation and controls. The DPV's are a pyrotechnically-actuated, straight-through, non-reclosing valve with a metal diaphragm seal. The use of a combination of SRVs and DPVs to accomplish the ADS function provides an improvement in ADS reliability against hypothetical common mode failures of otherwise non-diverse ADS components.

In the event of a LOCA, the ADS depressurizes the RPV sufficiently quickly to allow the gravity driven cooling system (GDCS) to inject coolant into the RPV, thereby keeping the core covered and maintaining the core temperature well below design limits. Following initial operation, the ADS maintains the reactor in a depressurized state without the need for AC or DC power. The SRVs and

DPVs are actuated in groups of valves at staggered times as the reactor undergoes depressurization. This minimizes reactor level swell during depressurization and minimizes the coolant inventory loss from the RPV.

4.7.2.2 Reactor core and fuel design

The ESBWR core configuration consists of 1020 bundles. The active fuel height is shorter than a typical BWR in order to reduce the pressure drop to augment the natural circulation. The rated core power is 4000 MW(th), which corresponds to a power density of 53.7 kW/l. Reactivity control is maintained by movement of control rods and the use of burnable poisons in the fuel. As backup, a standby liquid control system, which can inject a borated water solution into the reactor, is also available.

Control rod drive system

The control rod drive (CRD) system is composed of three major elements: the fine motion control rod drive (FMCRD) mechanisms; the hydraulic control unit (HCU) assemblies, and the control rod drive hydraulic (CRDH) subsystem.

The FMCRDs provide electric motor driven positioning for normal insertion and withdrawal of the control rods and hydraulic powered rapid control rod insertion (scram) in response to manual or automatic signals from the reactor protection system (RPS). Simultaneous with scram, the FMCRDs also provide electric motor driven run-in of all control rods as a means of rod insertion, that is diverse from the hydraulic powered scram. The hydraulic power required for scram is provided by high pressure water stored in individual hydraulic control unit (HCU) assemblies. Each HCU is designed to scram up to two control rods. The HCUs also provide the flow path for purge water to the associated drives during normal operation.

The CRDH subsystem supplies high pressure demineralized water which is regulated and distributed to provide charging of the HCU scram accumulators and purge water flow to the FMCRDs. The capacity of the pumps is sufficient to maintain RPV water level for small line break LOCAs. The CRD system is designed with the capability to provide makeup water to the RPV while at high pressure as long as AC power is available. If sensed reactor water level reaches Level 2 (approximately 8.6 m above the top of the active fuel), the CRD pumps run out in an effort to recover level.

The FMCRDs are mounted in housings welded into the RPV bottom head. Each FMCRD has a movable hollow piston tube that is coupled at its upper end, inside the reactor vessel, to the bottom of a control rod. The piston is designed such that it can be moved up or down, both in fine increments and continuously over its entire range, by a ball nut and ball screw driven at a nominal speed of 30 mm/s by the electric stepper motor. In response to a scram signal, the piston rapidly inserts the control rod into the core hydraulically, using stored energy in the HCU scram accumulator. The FMCRD design includes an electro-mechanical brake on the motor drive shaft and a ball check valve at the point of connection with the scram inlet line. These features prevent control rod ejection in the event of a failure of the scram insert line. Each HCU provides sufficient volume of water stored at high pressure in a pre-charged accumulator to scram two FMCRDs at any reactor pressure.

4.7.2.3 Fuel handling and transfer systems

The operating floor in the reactor building is serviced with a refueling platform. Figure 4.7-2 shows the operating floor layout.

The refueling platform is a dual robotic arm machine that spans the reactor vessel cavity and interim fuel storage (i.e. “buffer”) pool for the purpose of handling the fuel and performing other ancillary tasks. It is equipped with dual robotic arms for handling two fuel elements at a time. An auxiliary hoist is also provided. A programmed computer located above the refueling floor controls the platform’s operational movements. Mechanical stops and interlocks provide the necessary operational limits.

Transfer of spent fuel to the auxiliary fuel storage building, and new fuel to the operating floor, is achieved via an inclined fuel transfer system (IFTS) that handles two bundles at a time. This system is similar to that used in GE Mk III-style containment designs, with one key advantage – the transfer system is outside of containment. This allows fuel movement before and after an outage, which is a significant advantage.

The fuel buffer pool can store up to 336 new or spent fuel bundles, plus irradiated reactor components. With this capability, the sensitivity of fuel loading and unloading operations to the throughput of the IFTS is reduced from what otherwise would be the case.

Limited available water depth in the buffer pool requires the use of side-loading fuel storage racks. As required, the racks will be constructed of stainless steel with storage positions spaced to ensure that a full array of new or discharged fuel will remain subcritical by 5% Δk/k under all conditions.

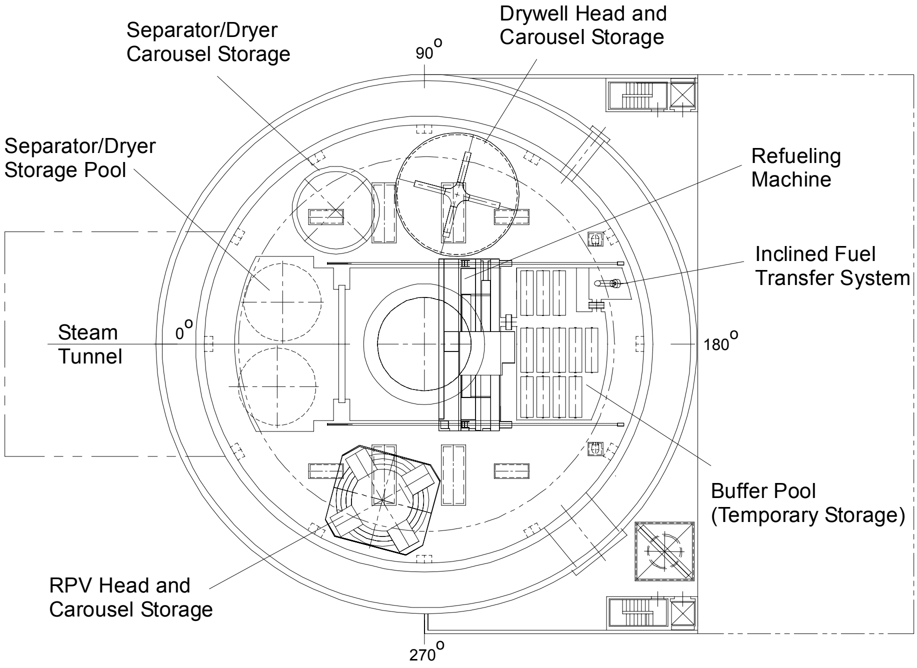

FIG. 4.7-2. ESBWR Operating Floor, Elev. 33600.

4.7.2.4 Primary components

Reactor pressure vessel

The ESBWR reactor pressure vessel (RPV) assembly consists of the pressure vessel, removable head, and its appurtenances, supports and insulation, and the reactor internals enclosed by the vessel (excluding the core, incore nuclear instrumentation, neutron sources, control rods, and control rod drives). The RPV instrumentation to monitor the conditions within the RPV is designed to cover the full range of reactor power operation. The RPV, together with its internals, provides guidance and support for FMCRDs. Details of the RPV and internals are discussed below.

The RPV is a vertical, cylindrical pressure vessel comprised of rings welded together, with a removable top head, head flanges, seals and bolting. The vessel also includes penetrations, nozzles, shroud support, and venturi shaped flow restrictors in the steam outlet nozzles.

The reactor vessel has a minimum inside diameter of 7.1 m (23.3 ft.), a wall thickness of about 182 mm (7.2 in.) with cladding, and a height of 27.6 m (90.3 ft.) from the inside of the bottom head (elevation zero) to the inside of the top head. The bottom of the active fuel location is 4.4 m (14.5 ft.) from elevation zero and the active core is 3.04 m (10.0 ft.) high. The relatively tall vessel permits natural circulation driving forces to produce abundant core coolant flow.

An increased internal flow path length, relative to forced circulation BWRs, is provided by a long "chimney" in the space that extends from the top of the core to the entrance to the steam separator assembly. The chimney and steam separator assembly are supported by a shroud assembly that extends to the top of the core. The resulting large RPV volume provides a substantial reservoir of water above the core, which insures the core remains covered following transients involving feedwater flow interruptions or loss-of-coolant-accidents (LOCAs). This gives an extended period of time during which automatic systems or plant operators can re-establish reactor inventory control using any of several normal, non-safety related systems capable of injecting water into the reactor. Timely initiation of these systems precludes the need for activation of emergency safety equipment.

The large RPV volume also reduces the rate at which reactor pressurization occurs if the reactor is suddenly isolated from its normal heat sink. If isolation should occur, reactor decay heat is rejected to an isolation condenser system (ICS) located within a large pool of water (the IC/PCC pool) positioned immediately above (and outside) the containment. The slower pressurization rate and the ICS eliminate the need to actuate relief valves, which would result in the discharge of RPV inventory to the suppression pool.

Reactor internals

The major reactor internal components include:

- core (fuel, channels, control rods and instrumentation),
- core support structures (shroud, shroud support, top guide, core plate, control rod guide tube and orificed fuel support),
- chimney and partitions,
- chimney head and steam separator assembly,
- steam dryer assembly,
- feedwater spargers,
- standby liquid control headers, spargers and piping assembly,
- incore guide tubes.

Except for the Zircaloy in the reactor core, these reactor internals are stress corrosion resistant stainless steels or other high alloy steels.

The fuel assemblies (including fuel rods and channels), control rods, chimney head and steam separator assembly, steam dryers and incore instrumentation assemblies are removable when the reactor vessel is opened for refueling or maintenance. In addition, the ESBWR internals are designed to be removable.

The RPV shroud support is designed to support the shroud and the components connected to the shroud. The RPV shroud support is a series of horizontal brackets welded to the vessel wall which support the weight of the steam separator, chimney, core plate, top guide and the fuel bundles.

Reactor recirculation pumps

The ESBWR operates with natural circulation of the reactor coolant. Thus, no recirculation pumps are provided.

4.7.2.5 Reactor auxiliary systems

Plant service water system (PSWS) - The PSWS rejects heat from non-safety related components in the reactor and turbine buildings to the environment. The PSWS consists of two independent and 100% redundant open loops continuously re-circulating water through the heat exchangers of the reactor component cooling water system (RCCW) and the turbine component cooling water system (TCCW). Heat is rejected to the environment by mechanical draft cooling towers.

Reactor component cooling water system (RCCW) - The RCCW cools non-safety related components in the reactor building and provides a barrier against potential radioactive contamination of the PSWS. The RCCW consists of two 100% capacity independent and redundant closed loops.

Makeup water system (MWS) - The MWS is designed to supply demineralized water to the various non-safety related systems that need demineralized water and provides water to the IC/PCC pools.

Condensate storage and transfer system (CSTS) - The CSTS is a non-safety related system that consist of two 100% pumps and lines taking suction from one storage tank that is the normal source of water for makeup to selected plant systems. The CSTS is also used for storage of excess condensate rejected from the condensate & feedwater systems and the condenser hotwell.

Chilled water system (CWS) - The CWS consists of two independent, non-safety related, sub-systems: the reactor building CWS and the turbine building CWS. The CWS provides chilled water to the cooling coils of air conditioning units and other coolers in the reactor and turbine buildings. Each subsystem consists of two 100% capacity, redundant, and independent loops with cross-ties between the chilled water piping.

4.7.2.6 Operating characteristics

The ESBWR relies extensively on the lessons learned from operating BWRs regarding natural circulation operation, especially the GKN Dodewaard natural circulation reactor. The ESBWR has been designed to maximize core flow and ensure that there are very large margins relative to operation in unstable flow regions. The core flow has been maximized by the following actions:

- Eliminating the flow restriction introduced by recirculation pumps (reduced pressure drop),
- Using shorter fuel (reduced pressure drop),
- Using a tall chimney above the core (increased driving head),
- Using improved steam separators (reduced pressure drop).

As shown in Figure 4.7-3, these design features result in an average core flow per bundle over three times greater than that of a forced circulation internal pump plant at similar bundle power. This allows the ESBWR to operate well away from regions of power/flow instability without the need for operating restrictions being imposed on the plant operator. As an added benefit, the use of natural circulation eliminates pumps, motors, controls, piping and many other components that could possibly fail and effect plant availability.

Similar to the Dodewaard plant, the ESBWR is extremely simple to operate during startup and normal operation, and also has a very gentle transient response. This is because of the large reactor vessel and steam and water inventory. A reactor isolation results in no loss of coolant inventory and no heating up of the containment or suppression pool.

The transient and accident responses of the plant are discussed in some detail in Ref. [1]. The extensive experience of startup and normal operation of BWRs with features common to the ESBWR, including nearly 30 years at Dodewaard (a natural circulation plant), provide high confidence in the design.

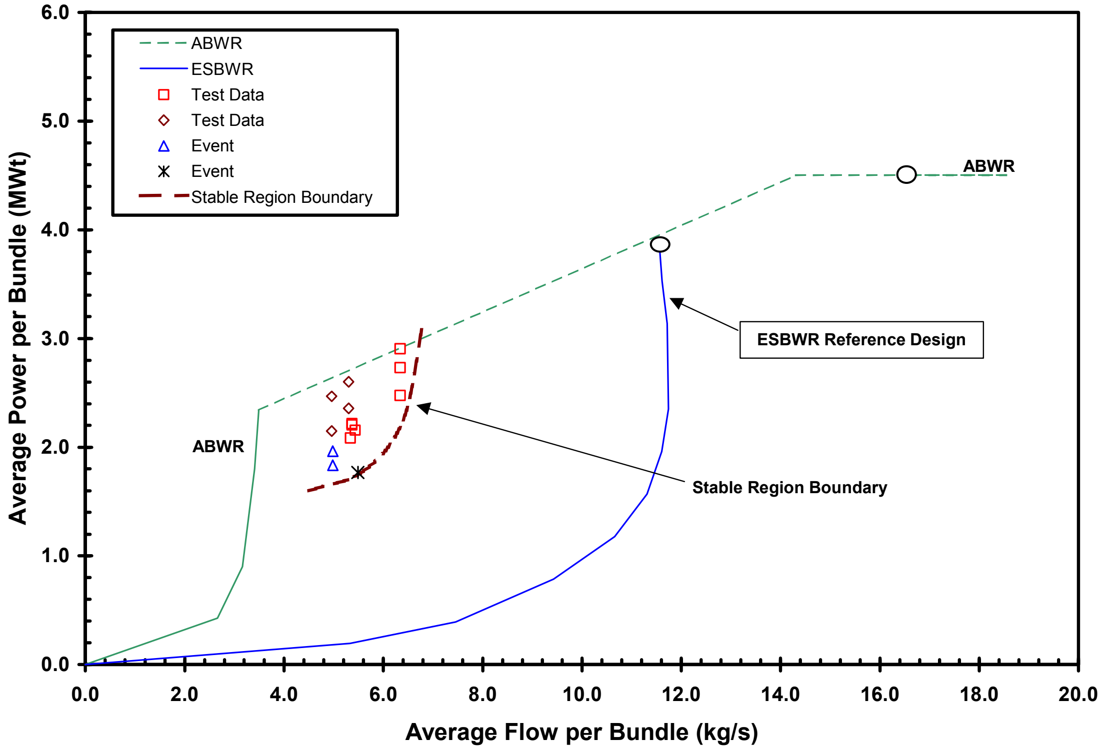

FIG. 4.7-3. Average core flow per bundle for ESBWR vs. forced convection BWRs.

4.7.3 Description of turbine generator plant systems

4.7.3.1 Turbine generator plant

The main turbine

The main turbine is a tandem compound, six flow, reheat type steam turbine with 14500 mm (57 inch) last stage buckets. There is one high pressure (HP) turbine and three low pressure (LP) turbines in series. The steam passes through an in-line high velocity moisture separator (HVS) and reheaters of advanced design prior to entering the LP turbines. Steam exhausted from the LP turbines is condensed and degassed in three main condensers arranged in series. The turbine uses steam at a pressure of 6.79 MPa (985 psia) from the reactor and rotates at 1500 RPM (for 50 Hz application) or 1800 RPM (for 60 Hz application). Steam is extracted from several stages of each turbine and is used to heat the feed-water.

Turbine overspeed protection system

In addition to the normal speed control function provided by the turbine control system, a separate turbine overspeed protection system is included to minimize the possibility of turbine failure and/or generation of high energy missiles.

Turbine gland seal system

The turbine gland seal system (TGSS) provides steam sealing to the labyrinth seals of the high pressure and low pressure turbines and to the stem seals of the turbine stop valves, control valves and bypass valves. The system prevents the escape of radioactive steam from the turbine shaft/casing penetrations and valve stems and prevents air in-leakage through sub-atmospheric turbine glands into the main condenser. The TGSS consists of a sealing steam pressure regulator, sealing steam header, a gland steam condenser, two full capacity exhaust blowers and associated piping, valves and instrumentation.

Turbine bypass system

A turbine bypass system (TBS) is provided which passes steam directly to the main condenser under the control of the pressure regulator. The TBS has the capability to shed 100% of the turbine generator rated load without reactor trip or operation of SRVs. The TBS does not serve or support any safety related function and has no safety design.

Main condenser

The main condenser is a multi-pressure, three-shell type, de-aerating condenser with each shell located directly beneath the respective low pressure turbines. It is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system. Each shell has tube bundles through which circulating cooling water flows. Condensing steam is collected in the condenser hot wells (the lower shell portion), providing suction to the condensate pumps. The main condenser has no safety design basis and does not serve or support any safety function.

Since the main condenser operates at a vacuum, any leakage is into the shell side. Tube side or circulating water in-leakage is detected by measuring the conductivity of sample water extracted beneath the tube bundles. In all operational modes, the condenser is at a vacuum so no radioactive releases can occur.

Non-condensable gases are removed from the power cycle by the main condenser evacuation system (MCES). The MCES removes power cycle non-condensable gases, including the hydrogen and oxygen produced by radiolysis of water in the reactor, and exhausts them to the offgas system (OGS) during plant power operation, or to the turbine building ventilation system exhaust during early plant startup. The MCES establishes and maintains a vacuum in the condenser by the use of steam jet air ejectors during power operation, and by a mechanical vacuum pump during early startup.

4.7.3.2 Condensate and feedwater systems

The condensate and feedwater system (C&FS) consists of the piping, valves, pumps, heat exchangers, controls and instrumentation and the associated equipment and subsystems which supply the reactor with heated feedwater in a closed steam cycle utilizing regenerative feedwater heating. The four condensate pumps take deaerated condensate from the condenser hot well and deliver it through the condensate demineralizer and through two strings of three low pressure feedwater heaters to a direct contact feedwater heater, which provides the equivalent performance of forward pumped heater drains. The direct contact feedwater heater receives condensate from the 5th and 6th stage feedwater heater drains, moisture separator drains and extraction steam from the LP turbines and provides 3–4 minutes of full power inventory to the vessel, which is sufficient to meet ESBWR transient feedwater flow requirements.

There are four variable-speed feedwater pumps, which take suction from the direct contact feedwater heater. Each feedwater pump is driven by a variable speed induction motor, with the combination sized to supply normal feedwater booster pump suction pressure. The feedwater pumps discharge directly to the suction side of corresponding reactor feedwater booster pumps. The feedwater booster pumps discharge through two high pressure feedwater heaters to the RPV.

4.7.3.3 Auxiliary systems

The radioactive waste management system consists of liquid, solid, detergent, and laundry waste management, and mobile systems. Liquid waste processing is done on a batch basis. Equipment drains and other low conductivity wastes are treated by filtration, UV/ozone, demineralization and transferred to the condensate storage tank for reuse. Floor drains and other high conductivity wastes are treated by filtration and ion exchange prior to being either discharged or recycled for reuse.

Detergent wastes of low activity are treated by filtration, sampled and released via the liquid discharge pathway. Chemical wastes are treated by filtration, sampled and released from the plant on a batch basis. Connections are provided for mobile processing systems that could be brought in to augment the installed waste processing capability. Connections for addition of a permanent evaporation subsystem are also provided in the event that site conditions warrant. Mixed waste will be segregated from the other types of radioactive waste for packaging.

The wet solid waste processing subsystem consists of a builtin dewatering station. A high integrity container (HIC) is filled with either sludge from the phase separator or bead resin from the spent resin tanks. Spent cartridge filters may also be placed in the HIC.

Dry wastes consists of air filters, miscellaneous paper, rags, etc., from contaminated areas; contaminated clothing, tools and equipment parts that cannot be effectively decontaminated; and solid laboratory wastes. The activity of much of this waste is low enough to permit handling by contact. These wastes are collected in containers located in appropriate areas throughout the plant. The filled containers are sealed and moved to an enclosed, access-controlled area for temporary storage.

4.7.4 Instrumentation and control systems

4.7.4.1 Design concepts, including control room

The ESBWR control and instrument systems provide manual and automatic means to control plant operations and initiate protective actions should plant upset conditions occur. The ESBWR utilizes digital controllers, interfacing with plant equipment, sensors and operator controls through a multiplexing system, for signal transmission to achieve these functions. The key distinguishing simplification features for plant control and monitoring include:

- Enhanced man-machine interface design
- Automated plant operations
- Simplified neutron monitoring system
- Reduction in number of nuclear boiler instruments
- Fault tolerant safety system logic and control
- Standardized digital control and measurement
- Multiplexing of plant control signals

Multiplexed signal transmission using high speed fiber optic data links is combined with digital technology to integrate control and data acquisition for both reactor and turbine plants. Multiplexing significantly reduces the quantities of control cables, which need to be installed during construction, thereby reducing the construction cost, and facilitates automation of plant operations.

Performance monitoring and control, and power generator control subsystem functions are provided by the process computer system (PCS) to support efficient plant operation and automation.

The main control room (MCR) panels consist of an integrated set of operator interface panels (e.g. main control console, large display panel). The safety-related panels are seismically qualified and provide grounding, electrical independence and physical separation between safety divisions and non-safety related components and wiring.

The MCR panels and other MCR operator interfaces are designed to provide the operator with information and controls needed to safely operate the plant in all operating modes, including startup, power operation, refueling, shutdown, and cold shutdown. Human factors engineering principles have been incorporated into all aspects of the ESBWR MCR design.

The liquid and solid radwaste systems are operated from control panels in the radwaste control room. Programmable controllers are used in this application.

4.7.4.2 Reactor protection system and other safety systems

The safety system logic and control (SSLC) provides a centralized facility of implementing safety related logic functions. The SSLC is configured as a four-division data acquisition and control system, with each division containing an independent set of microprocessor based software controlled logic processors.

The reactor protection system (RPS) is an overall complex of instrument channels, trip logic, trip actuators, manual controls, and scram logic circuitry that initiates the rapid insertion of control rods by hydraulic force to scram the reactor when unsafe conditions are detected. The RPS uses the functions of the essential multiplexing subsystem (EMS) and the SSLC system to perform its functions.

The remote shutdown system (RSS) provides the means to safely shutdown the reactor from outside the main control room. The RSS provides remote manual control to the systems necessary to: (a) achieve prompt hot shutdown of the reactor after a scram, (b) achieve subsequent cold shutdown of the reactor, and (c) maintain safe conditions during shutdown.

The standby liquid control system (SLCS) provides an alternate method of reactor shutdown from full power to cold subcritical by the injection of a neutron absorbing solution to the RPV. SLCS initiates automatically as required to mitigate an anticipated transient without scram (ATWS).

The feedwater control system (FWCS) controls the flow of feedwater into the RPV to maintain the water level in the vessel within predetermined limits during all plant operating modes.

The neutron monitoring system (NMS) provides indication of neutron flux in the core in all modes of reactor operation. The safety related NMS functions are the startup range neutron monitor (SRNM), the local power range monitor (LPRM), and the average power range monitor (APRM). The non-safety related subsystem is the automated fixed in core probe (AFIP). The LPRMs and APRMs make up the power range neutron monitor (PRNM) subsystem.

The NMS provides signals to the RPS, the rod control and information system (RC&IS), and the process computer system. The NMS provides trip signals to the RPS to scram the reactor on high neutron flux or high thermal power. In the startup range, the SRNM provides a trip signal for excessively short reactor periods to mitigate neutron flux excursions.

4.7.5 Electrical systems

4.7.5.1 Operational power supply systems

On-site power is supplied from the plant turbine generator, utility power grid, or an off-site power source, depending on the plant operating status. During normal operation, plant loads are supplied from the main generator through the unit auxiliary transformers. A generator breaker allows the unit auxiliary transformers to stay connected to the grid to supply loads by back feeding from the switchyard when the turbine is not on-line.

Individual voltage regulating transformers supply 120 V AC, non-safety related, control and instrument power.

Standby AC power supply

The non-safety related Standby AC power supply consists of two diesel generators (DG). Each unit provides 6.6 KV AC power to one of the two load groups whenever the main turbine generator and the normal preferred off-site power source are not operating. When operating, the standby AC power supply provides power to non-safety related investment protection loads but can be connected to power safety related loads. In ESBWR all Class 1E loads are supplied power by eight Class 1E 125 V DC batteries and eight Class 1E inverters.

Direct current power supply

Non-Class 1E DC power is supplied through four non-Class 1E 600 V AC motor control centers (MCCs). Each of the two load groups receives power from two of the non-Class 1E MCCs. One MCC in each group provides power to a 250 V DC bus through a battery charger. A 250 V DC station battery provides backup to the supply from the battery charger. A 250 V DC station battery provides backup to the supply from the battery charger. The 250 V DC batteries supply the DC motors that protect rotating machinery in case of plant power loss and supply the large inverters that power the plant's normal instrumentation and control loads.

During a loss of off-site power, the non-Class 1E systems are powered automatically from the standby diesel generators. If these are not available, power to essential loads is provided by the 125 V DC and 250 V DC station batteries.

Instrument and control power supply

The instrument and control power supply provides 120 V AC single phase power to instrument and control loads that do not require an uninterruptable power source.

4.7.5.2 Safety related systems

Direct current power supply

The Class 1E DC power supply provides power to the Class 1E vital AC buses through inverters, and to 125 V DC loads required for safe shutdown. Each of the four divisions of class 1E DC power is separate and independent. Each division has a 125 V DC battery and a battery charger fed from its divisional 600 V AC MCC. This system is designed so that no single failure in any division of the 125 V DC system will prevent safe shutdown of the plant.

During a total loss of off-site power, the Class 1E system is powered automatically from two non-Class 1E standby diesel generators. If these are not available, each division of Class 1E isolates itself from the non-Class 1E system, and power to safety related loads is provided uninterrupted by the Class 1E batteries. The batteries are sized to power safety related loads for a 72-hour period.

Vital (uninterruptable) power supply

The Class 1E vital AC power supply provides redundant, reliable power to the safety logic and control functions during normal, upset and accident conditions.

Each of the four divisions of this class 1E vital AC power is separate and independent. Each division is powered from an inverter supplied from a Class 1E DC bus. The DC bus receives its power from a divisional battery charger and battery. Provision is made for automatic switching to an alternate Class 1E non vital supply in case of failure of the inverter.

4.7.6 Safety concept

4.7.6.1 Safety requirements and design philosophy

The basic ESBWR safety design philosophy is built on utilization of inherent margins (e.g. larger volumes and water inventory) to eliminate system challenges. An example of this philosophy is that during reactor isolation, no SRV shall actuate. The first line of defense is to enhance the normal operating system's ability to handle transients and accidents through such design features as adjustable speed, motor driven, feedwater pumps and higher capacity CRD pumps with backup power (6.6 kV plant investment protection buses). As a second line of defense, passive safety related systems are used in the design to provide confidence in the plant's ability to handle transients and accidents.

The plant also retains several motor driven (non-safety) systems to handle transients and accidents. As well, all safety related systems are designed such that no operator actions are needed to maintain safe, stable, conditions for 72 hours following a design basis accident. Descriptions of some important passive safety related systems are provided in the following section.

4.7.6.2 Safety systems and features (active, passive and inherent)

Isolation condenser system (ICS)

The isolation condenser system removes decay heat after any reactor isolation during power operations. Decay heat removal limits further increases in steam pressure and keeps the RPV pressure below the SRV set point. The ICS consists of four independent loops, each containing a heat exchanger that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC pool, which is vented to the atmosphere. The arrangement of the IC heat exchanger is shown in Figure 4.7-4.

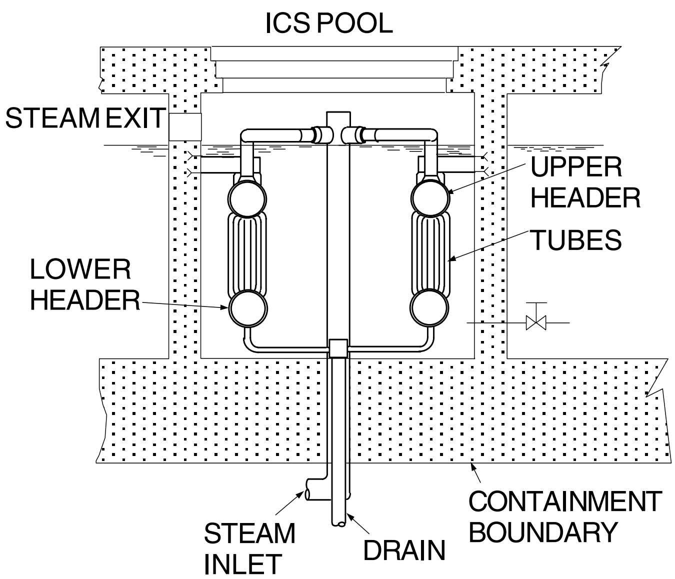

FIG. 4.7-4. Isolation condenser arrangement.

The ICS is initiated automatically by any of the following signals: high reactor pressure, MSIV closure, or an RPV water Level 2 signal. To start an IC into operation, the IC condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam water interface in the IC tube bundle moves downward below the lower headers. The ICS can also be initiated manually by the operator from the MCR by opening the IC condensate return valve.

The IC pool has an installed capacity that provides ~72 hours of reactor decay heat removal capability. The heat rejection process can be continued indefinitely by replenishing the IC pool inventory. The

ICS passively removes sensible and core decay heat from the reactor when the normal heat removal system is unavailable. Heat transfer from the IC tubes to the surrounding IC pool water is accomplished by natural convection, and no forced circulation equipment is required.

Emergency core cooling — gravity driven cooling system (GDCS)

The gravity driven cooling system (GDCS), in conjunction with the automatic depressurization system (ADS), comprise the emergency core cooling system (ECCS) for ESBWR. Following a confirmed RPV water Level 1 signal and depressurization of the reactor to near-ambient pressure conditions by the ADS, the GDCS will inject large amounts of cooling water into the reactor. The cooling water flows to the RPV through simple, passive, gravity-draining.

The GDCS is composed of four identical, safety-related, divisions. For "short term" cooling needs, each division takes suction from three independent GDCS pools positioned in the upper elevations of the containment (Figures 4.7-1 and 4.7-8). Flow from each division is controlled by pyrotechnic-type ECCS injection valves, which remain open after initial actuation. Each division of the "short term" subystem feeds two GDCS injection nozzles on the RPV (8 total).

If a pipe break at lower RPV elevations results in the total draindown of the GDCS pools (i.e., a bottom drainline break), "long term" cooling needs are provided by a second GDCS subsystem fed by water from the suppression pool. Pyrotechnic-type ECCS injection valves are also used in this subsystem, which feeds one RPV injection nozzle per division (4 total). These nozzles are placed at a lower elevation on the RPV than those of the "short term" subsystem.

In the event of a postulated severe accident that results in a core melt, with the molten core reaching the lower drywell region, the three upper GDCS pools have sufficient inventory to flood the lower drywell cavity to a level equal to the top of the active fuel.

The GDCS is completely automatic in actuation and operation. The ability to actuate the system manually is provided as a backup, but the operator cannot close any valves in the system.

Passive containment cooling system (PCCS)

The PCCS is a passive system which removes the decay heat and maintains the containment within its pressure limits for design basis accidents such as a LOCA. It consists of four low pressure, totally independent, loops, each containing a steam condenser in a pool of water (Figure 4.7-5), with the steam inlet coming from the drywell area surrounding the RPV.

The steam condenser condenses steam on the tube side and transfers heat to the water in the IC/PCC pool. The IC/PCC pool is vented to the atmosphere. Each PCCS condenser is located in a sub-compartment of the IC/PCC pool, and all pool sub-compartments communicate at their lower elevations. This allows full use of the collective water inventory, independent of the operational status of any given PCCS loop.

The PCCS loops are driven by the pressure difference created between the containment drywell and the suppression pool during a LOCA. PCCS operation requires no sensing, control, logic or power actuated devices for operation. Together with the pressure suppression containment system, the four PCCS condensers limit containment pressure to less than the design pressure for at least 72 hours after a LOCA, without inventory makeup to the IC/PCC pool.

The PCCS condensers are a closed loop extension of the containment pressure boundary and are designed for twice the containment design pressure. Since there are no containment isolation valves between the PCCS condensers and the drywell, they are always in "ready standby" mode.

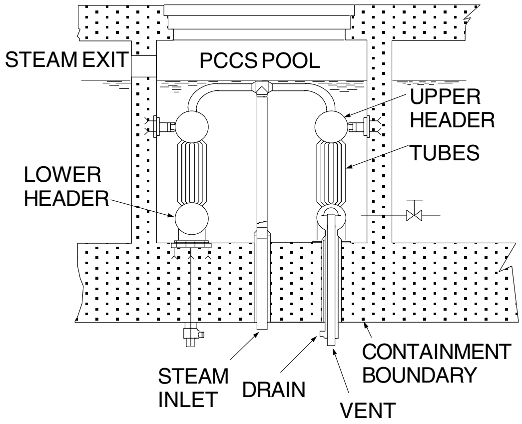

FIG. 4.7-5. Passive containment cooling condenser arrangement

TABLE 4.7-I. KEY ESBWR FEATURES FOR PREVENTION AND MITIGATION OF SEVERE ACCIDENTS

Design Feature	**Purpose/Description**
Compact containment design with minimum penetrations. Lower drywell kept dry (Mitigation)	Containment isolation with minimum leakage. High retention of aerosols. Fuel coolant interactions minimized.
Isolation condenser system (Prevention)	Controls reactor pressure. First line of defense against accidents.
Diverse automatic depressurization system (Prevention)	Depressurizes reactor pressure vessel and prevents high pressure core melt. Minimizes probability of direct containment heating.
Passive containment cooling system (Prevention and Mitigation)	Provides long term containment cooling. Keeps pressure within design limits.
PCC heat exchanges (Mitigation)	Filter aerosols - minimize offsite dose.
Suppression pool and airspace (Prevention & Mitigation)	Suppression pool is heat sink. Scrubs aerosols. Airspace volume is sized for 100% metal water reaction.
GDCS in wet well configuration (Prevention & Mitigation)	Increases airspace volume to handle non-condensable gas release in severe accident situations.
Core catcher (Mitigation)	Retention of molten core. Core catcher prevents basemat erosion and melt through. Prevents core-concrete interaction.
Lower drywell configuration (Mitigation)	Lower drywell floor provides sufficient spreading area for cooling of a molten core.
Lower drywell flooder system (Mitigation)	Provides external vessel cooling (in-vessel retention) and additional cooling for corium on the floor.
Inerted containment (Prevention & Mitigation)	Prevents hydrogen detonation.
Recombiners / igniters (Prevention & Mitigation)	Prevents hydrogen and/or oxygen combustion and detonation.
Containment overpressure protection system (Mitigation)	An optional system that provides additional defense in depth.

4.7.6.3 Severe accidents (Beyond design basis accidents)

The ESBWR design philosophy on plant safety is one of “prevention and mitigation through simplification”. Prevention is achieved by utilizing a systematic design approach that provides simplified but diverse and redundant systems or components. Mitigation is achieved in two ways. First, by assuring the integrity of the containment under severe accident conditions. Second, by providing adequate fission product control so as to minimize offsite dose and consequences to the general population. Key ESBWR features with respect to prevention and mitigation of severe accidents are summarized in Table 4.7-I.

4.7.7. Plant layout

4.7.7.1 Buildings and structures, including plot plan

The plot plan, showing the general layout of the ESBWR buildings, is depicted in Figure 4.7-6. The principal plant structures of the ESBWR are: the reactor building, the control building, the turbine building, the radwaste building, and the electrical building.

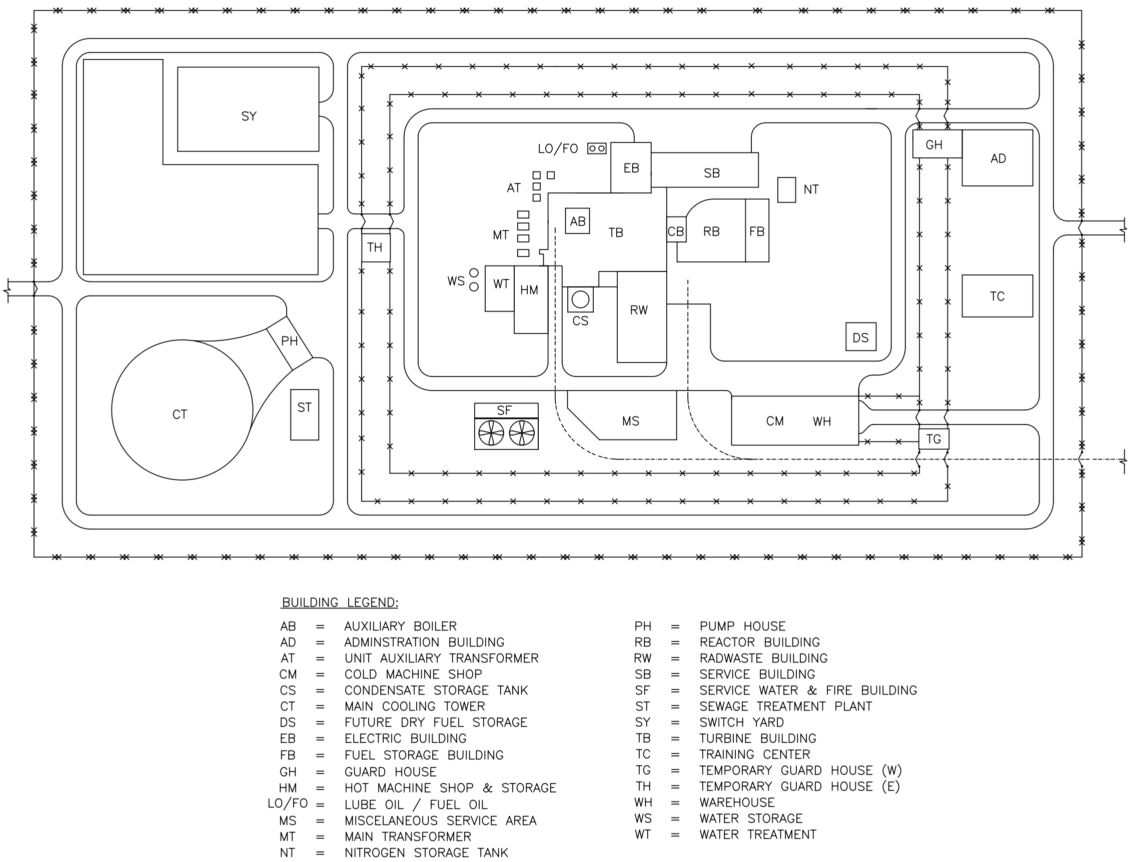

FIG. 4.7-6. ESBWR Plant Layout, Site Plot Plan

Design requirements

Earthquake - The reactor building seismic design is based on the US NRC Regulatory Guide 1.60 spectrum for a wide range of soil conditions. European seismic requirements are expected to be less stringent and therefore bounded by the ESBWR design.

Aircraft crash - The reactor building will be designed to the applicable requirements.

Explosion pressure wave - The reactor building will be designed to the applicable requirements.

Internal hazards - Internal loads from hazards will be included in the containment design.

Physical separation aspects - Safety grade systems are protected by physical, or spatial, separation wherever possible. When spatial separation is not possible, physical barriers will be used to provide the required equivalent separation.

Radiation protection aspects (accessibility, shielding, ventilation) - The reactor building layout separates controlled from non-controlled areas by separation of the respective equipment.

4.7.7.2 Reactor building

Most of the components, equipment and systems providing safety related functions in the ESBWR are housed in the reactor building, the main steam tunnel, or the auxiliary fuel storage building. This includes the reactor containment, the refueling area, temporary spent fuel storage areas, and support equipment. Non-safety related systems are typically placed in buildings adjacent to this envelope. Figures 4.7-7 through 4.7-9 show the major features of the ESBWR reactor building.

The reactor building is a Seismic Category I structure. As shown in the above figures, it is enclosed by a secondary shell providing protection of electrical and mechanical penetrations in the reactor building walls. The reactor building surrounds the cylindrical reinforced concrete containment vessel (RCCV). Both structures are located on a common basemat. The reactor building outer walls are reinforced concrete shear walls and the building is partially embedded.

4.7.7.3 Containment

The ESBWR containment structure is a reinforced concrete cylindrical structure that encloses the reactor pressure vessel (RPV) and its related systems and components. The containment is divided into a drywell region and a suppression chamber region, with a vent system connecting the two. The details of the containment structure are shown in Figures 4.7-1, 4.7-7, and 4.7-8.

The drywell region is a leak tight gas space surrounding the RPV and reactor coolant pressure boundary. It provides confinement of radioactive fission products, steam and water released in the unlikely event of a LOCA. The containment is designed to direct the fission products, steam and water released during a LOCA to the suppression pool via the vent system.

The suppression chamber region consists of the suppression pool and the gas space above it. The suppression pool is a large body of water that absorbs the LOCA energy by condensing steam from safety/relief valve discharges and RPV blowdown energy. The gas space above the suppression pool is leak tight and sized to collect and retain the drywell gases following a pipe break in the drywell, without exceeding the containment design pressure.

Three enclosed pools of water sit above the suppression pool gas space, at the periphery of the upper drywell. These pools are part of a gravity driven cooling system (GDCS), which can supply makeup water to the RPV in the event of a LOCA. The gas space in these pools is connected to the suppression pool gas space so, as they empty out, the available expansion volume for non-condensibles discharged to the suppression pool is increased.

4.7.7.4 Auxiliary fuel storage building

The auxiliary fuel building is located adjacent to the reactor building, on the opposite side from the turbine building. Facilities for wet, long term storage of spent fuel are provided in this building, as well as areas for storage, inspection and staging of new fuel prior to insertion into the reactor. The nominal spent fuel storage capacity is 2160 bundles, which corresponds to approximately one full core plus six reloads. However, this capacity can readily be adjusted to the operator’s needs since the pool is at grade level. Fuel storage basket design for this building will be conventional high density racks.

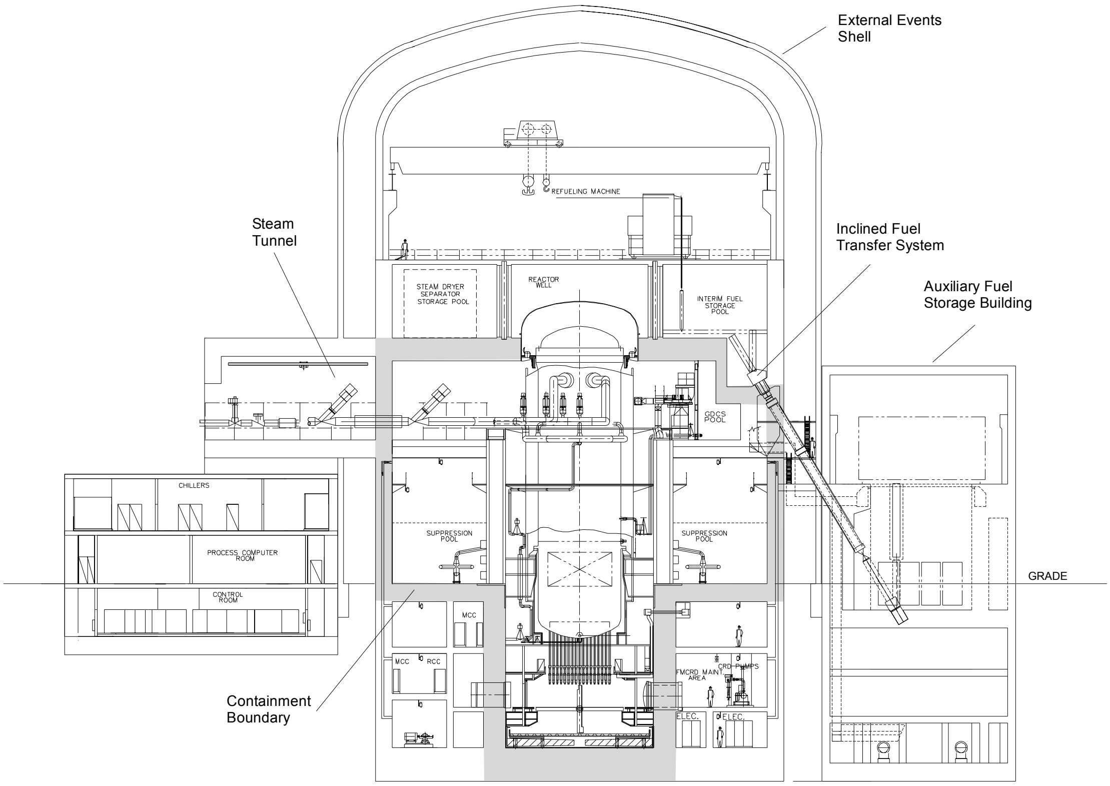

FIG. 4.7-7. ESBWR reactor building arrangement, Section A-A, 0º - 180º.

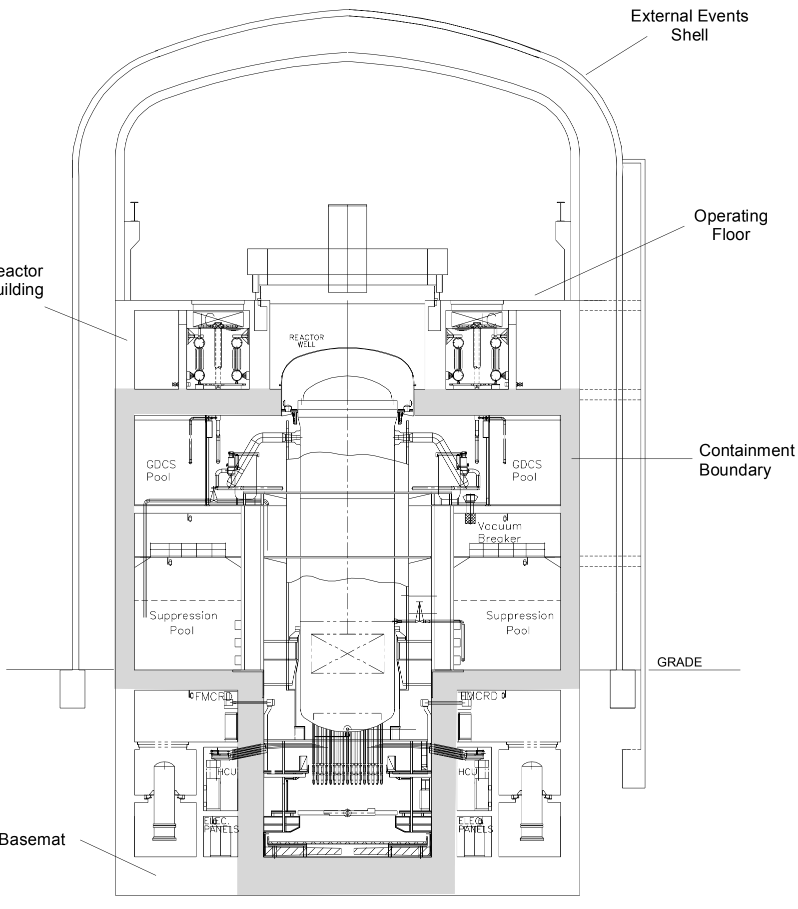

FIG. 4.7-8. ESBWR reactor building arrangement, Section B-B, 90º – 270º.

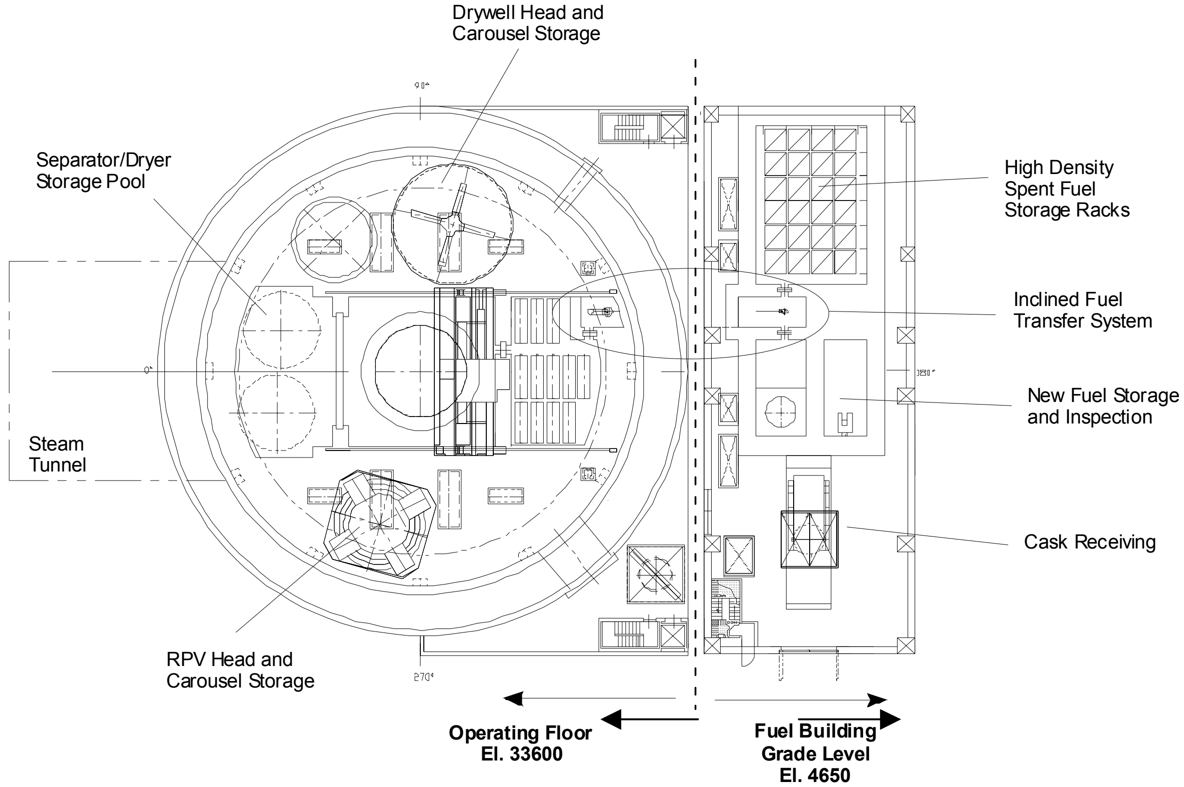

FIG. 4.7-9. ESBWR reactor building arrangement, operating floor, elev. 33600 mm.

4.7.7.5 Turbine building

The non-safety related turbine building houses equipment associated with the main turbine and generator and their auxiliary systems. Equipment including the condensate purification system, the process offgas treatment system, and the reactor component cooling water (RCCW) system are located in this building. It is a reinforced concrete structure up to the turbine operating deck; above that the building is constructed of steel frame and metal siding. Shielding is provided for the turbine on the operating deck and the turbine generator and condenser are supported on spring type foundations.

4.7.7.6 Other buildings

Control building

The control building is located under the steam tunnel, between the reactor building and turbine building. It houses the main control room and safety related instrumentation plus the associated logic control panels.

Radwaste building

The radwaste building houses equipment for collecting and processing solid and liquid radioactive waste generated by the plant. The structure up to grade elevation is reinforced concrete and has a structural steel framework with metal siding and a metal roof above that. The below grade portion of the building is designed to the requirements of Regulatory Guide 1.143, and the balance of the structure is classified non-seismic.

Electrical building

The non-safety related electrical building houses the two non-safety related standby diesel generators and their associated auxiliary equipment.

4.7.8 Technical data

<u>General plant data</u>		
Power plant output, gross	1390	MW(e)
Power plant output, net	1333	MW(e)
Reactor thermal output	4000	MWt
Power plant efficiency, net	33.3	%
Cooling water temperature	38	°C
<u>Nuclear steam supply system</u>		
Coolant loops	1	
Reactor pressure vessel free volume	959	m^3
Steam flow rate at nominal conditions [7790 t/hr]	2164	kg/s
Feedwater flow rate at nominal conditions [7780 t/hr]	2161	kg/s
<u>Reactor coolant system</u>		
Primary coolant flow rate	12,064	kg/s
Reactor operating pressure	7.17	MPa
Steam temperature/pressure	287.7 / 7.17	°C/MPa
Feedwater temperature	215.8	°C
Core coolant inlet temperature	276.2	°C
Core coolant outlet temperature	287.7	°C
Mean temperature rise across core	11.5	°C
<u>Reactor core</u>		
Active core height	3.048	m
Equivalent core diameter	5.58	m
Heat transfer surface in the core	8763	m^2
Average linear heat rate	14.7	kW/m
Fuel weight	146.6	t U
Average fuel power density	27.3	kW/kg U
Average core power density	53.7	kW/l
Thermal heat flux, F_q	456	kW/m^2
Enthalpy rise, F_H	(later)	

Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	3780 max.	mm
Rod array lattice	10x10 (square)	
Number of fuel assemblies	1020	
Number of fuel rods/assembly	92 (full + partial length)	
Number of spacers	6	
Enrichment (range) of first core, average	3.61	Wt%
Enrichment of reload fuel at equilibrium core	(later)	Wt%
Operating cycle length (fuel cycle length)	12-24	months
Average discharge burnup of fuel	41.8	GWd/t
Cladding tube material	annealed, recrystallised Zr 2	
Cladding tube wall thickness	0.6604	mm
Outer diameter of fuel rods	10.26	mm
Fuel channel/box; material	Zr-4	
Overall weight of assembly, including box	238	kg
Uranium weight/assembly	144	kg
Active length of fuel rods	3048 (max.)	mm
Burnable absorber, strategy/material	Gd_2O_3 mixed with fuel	
Number of control rods	(later)	
Absorber material	B_4C/hafnium	
Drive mechanism	electro-mechanical	
Positioning rate	30	mm/*s*
Soluble neutron absorber	Sodium Pentaborate	
<u>Reactor pressure vessel</u>		
Inner diameter of cylindrical shell	7100	mm
Wall thickness of cylindrical shell	182	mm
Total height, inside	27,600	mm
Base material: cylindrical shell	low-alloy carbon steel	
RPV head	[to ASTM A533, grade B, ASTM A508, class 3 or equiv.	
lining	SS cladding	
Design pressure/temperature	8.62 / 300	MPa/°C
Transport weight [lower part, including head]	853	t
RPV head	100	t

Reactor recirculation pump

Type	not applicable	
Number	0	
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	Single wall, pressure suppression	
Overall form (spherical/cyl.)	Cylindrical, reinforced concrete with steel liner.	
Dimensions (outer dia./height): lower drywell	13.2 / 17	m
upper drywell	33.5 / 7.1	m
wetwell	33.5 / 11.3	m
Design pressure/temperature	414 / 171	kPa(abs)/°C
Design leakage rate	0.5	vol%/day
Is secondary containment provided?	Partial	

Reactor auxiliary systems

Reactor water cleanup/Shutdown cooling, capacity	.0011	kg/s
filter type	mixed bed type	
Residual heat removal, at high pressure	132.3	kg/s
at low pressure	132.3	kg/s
Coolant injection, at high pressure	n/a	kg/s
at low pressure	n/a	kg/s

Power supply systems

	# of units		
Main transformer [1 phase] each	three	529	MVA
	[plus a spare]	1587	MVA total
Unit transformers	two	103	MVA
		dual voltage secondaries	
Reserve transformers	one	121	kVA
		dual voltage secondaries	
Medium voltage busbars	four	11.5	kV
	four	6.6	kV
Low voltage busbar systems	four	600	VAC 1E
Standby diesel generating units	two	6.6	kV
		6.8 / 8	MW(e)/MVA
Diesel backed busbar systems	four	6.6 kV bussbars	
DC distributions	eight	125	VDC 1E
		four normal 2 hour, or 72 hour coping	
	two	125	VDCnon 1E
	two	250	VDCnon 1E
Battery backed busbar systems	eight	220	VAC 1E
	[four normal 2 hour, or 72 hour coping]		
	two	220	VAC non 1E

Turbine plant

Turbines per reactor	one	
Type of turbine(s)	tandem compound, six flow	
Turbine sections per unit (e.g. HP/LP/LP)	one HP / three LP	
last stage blade length (LSB), nominal	1450	mm
Turbine speed	1500	rpm
Overall length of turbine unit	(later)	m
Overall width of turbine unit	(later)	m
HP inlet pressure/temperature	6.8 / 287	MPa/°C

Generator

Type	4-pole, 3-phase, turbo-generator	
Rated power [gross/net]	1390 /1333	MW(e)
Active power	1580	MVA
Voltage, nominal	24 - 27	kV
Frequency	50	Hz
Total generator mass, including exciter	(later)	t
Overall length of generator	(later)	m

Condenser

Type	series arrangement, three shells	
Heat sink	natural draft cooling tower	
Number/type of tubes	(later) / titanium or stainless steel	
Heat transfer area, nominal	(later)	m^2
Cooling water flow rate, nominal	(later)	m^3/s
Cooling water temperature (max)	38	°C
Condenser pressure, nominal		
LP cond / HP cond	5.4 / 9.5	kPa

Condensate pumps

Number		4
Flow rate	1425	kg/s
Pump head	(later)	MPa
Temperature	42	°C
Pump speed	(later)	rpm

Condensate clean-up system

Full flow/part flow	full condensate flow
Filter type	(later)

Direct contact feedwater heater tank

Volume	500	m^3
Pressure/temperature	0.321 / 136	MPa/°C

Feedwater booster pumps

Number	4	
Flow rate	2163	kg/s
Pump head	(later)	MPa
Feed pump power	(later)	MW
Feedwater temperature (final)	216	°C
Pump speed	(later)	rpm

Condensate and feedwater heaters

Heating stages	[LP / HP]	4 / 2

4.7.9 Measures to enhance economy and maintainability

Key Attribute	Elements of Attribute	Design Features
Simplification	Reduced systems	Passive safety systems
	Reduced structures	Smaller safety related structures
	Simpler operation	Eliminate recirculation pumps and associated support components
Operation flexibility	Performance margins	Large vessel
		Lower pressurization rate
		Passive systems
		Margins for core coverage
	Lower demand upon operator	Simple decay heat removal
	No immediate action required	Large passive coolant inventory
Economics	Low plant cost	Reduced materials
		Reduced equipment especially active components
		Reduced buildings
		Located most non-safety equipment in non-safety buildings
	Low development cost	ABWR & SBWR features used
	Licensing and first plant cost	New components and systems tested
Construction Period	Reduced construction time	Design adaptable to modularization
		Parallel construction of reactor building and fuel building
Maintenance	Reduced maintenance cost	Simpler systems
		Fewer active systems and components
	Ease of maintenance and radiation exposure	Less maintenance on control rod drives
		Eliminated maintenance on recirculation system
		Reduced maintenance on ECCS components
Plant availability	Reduced outage length	Design for servicing
		Dual robotic refueling
Power generation cost	Design for higher availability	See above
	Design for reduced O&M	See above

4.7.10 Project status and planned schedule

The ESBWR program is based on the earlier SBWR program, which was sponsored by the US Department of Energy (DOE). The ESBWR program was started in 1993 to improve the economics of the SBWR design. A multi-year, four-phase program has been defined to complete the technology, develop a detailed design, and secure certification with regulatory bodies. Evaluations of the overall design show that the plant has been considerably simplified and that the overall material quantities, on a per kWe basis, are significantly lower than those for the SBWR design and other GE designs.

The design and technology program involves several utilities, design organizations and research groups in seven countries. Overall design leadership is provided by General Electric Company (GE - USA) and overall program guidance is provided by an ESBWR steering committee. In mid-2002, the technology base was submitted to the U.S Nuclear Regulatory Commission for review with the objective of obtaining closure of all technology issues in 2003. This is a first and necessary step toward obtaining NRC design certification.

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4.8 KSNP$^+$ (KOREA HYDRO AND NUCLEAR POWER COMPANY, REP. OF KOREA)

4.8.1 Introduction

As the international economic environment has recently entered into an era of increased competition due to the initiation of the WTO regime and the complete opening of many countries' markets, so too has the international nuclear industry been inevitably exposed to strong competition. To positively cope with these changes in the economic environment, it was inevitably required that the design concept of the existing KSNP be re-established based on Korea's enhanced technical capabilities and accumulated construction and operation experience, and be re-configured as a new evolutionary nuclear power plant model; that is, it is desirable to develop a more internationally marketable "improved KSNP (KSNP$^+$)", rather than focusing on partial design improvements of the existing KSNP series, and enhancing the safety and economy of the KSNP design by incorporating reformative and comprehensive improvements.

The KSNP$^+$ design improvement program (KSNP$^+$ Program) is currently in progress, segmented in three phases. Phase I of the KSNP$^+$ Program started in January 1998 and was completed in January 1999. In Phase I of the KSNP$^+$ Program, 103 items were selected for detailed examination after a preliminary review of 323 items proposed by the operators, constructors, vendors, and designers of the existing nuclear power plants. 87 items were finally selected for design improvement to be applied to the KSNP$^+$ design after detailed reviews of the technological and economical aspects of each item.

Phase II of the KSNP$^+$ Program, from October 1999 to October 2001 proceeded to refine and firm up the improvements proposed during Phase I. Comprehensive design verifications for the results of Phase I were performed on the features of licensing, functional requirements, component design details, constructibility, operability and maintainability. The following goals were set for Phase II:

- Improve the international marketability and competitiveness of the KSNP$^+$ design through reduction in initial investment and operating costs;
- Improve constructibility, operability and maintainability.

13 items for design improvement were additionally reviewed during Phase II.

Phase III of the KSNP$^+$ Program, which began in February of 2001, endeavors to realize the KSNP$^+$ design through the construction of Shin-Kori Units 1&2, which will represent the first-fruits of the KSNP$^+$ Program, and are expected to be among the safest, most economical and advanced nuclear power plants in the world.

The KSNP$^+$ design incorporates features to improve safety, technology, economics, operability, and maintainability, when compared with the KSNP design. Obviously, the KSNP$^+$ aims at both enhanced safety and economic competitiveness. From the point of view of Probabilistic Safety Assessment (PSA), KSNP$^+$ will have an approximately 11% lower probability of core damage versus the KSNP design. The economic goal of KSNP$^+$, to secure a significant cost advantage over competitive energy sources such as coal-fired power generation, is considered achievable via high performance during operation and cost savings in construction.

4.8.2 Description of the nuclear systems

4.8.2.1 Primary circuit and its main characteristics

The primary loop configuration of the KSNP$^+$ is identical to that of the KSNP, in that it has two reactor coolant loops. The nuclear steam supply system is designed to operate at a maximum core thermal output of approximately 2,815 MWt to produce an electric power output of 1,050 MWe in the turbine/generator system. The major components of the primary circuit are the reactor vessel; two reactor coolant loops, each containing one hot leg and two cold legs; one steam generator (SG) and two reactor coolant pumps (RCPs); and a pressurizer (PZR) connected to one of the hot legs. All components are located inside the containment vessel. The two SGs and the four RCPs are arranged symmetrically. The steam generators are located at a higher elevation than the reactor vessel for natural circulation purposes. For venting and draining purposes, the elevation of the PZR and the surge line is higher than that of reactor coolant piping. The RCS diagram is shown in Figure 4.8-1.

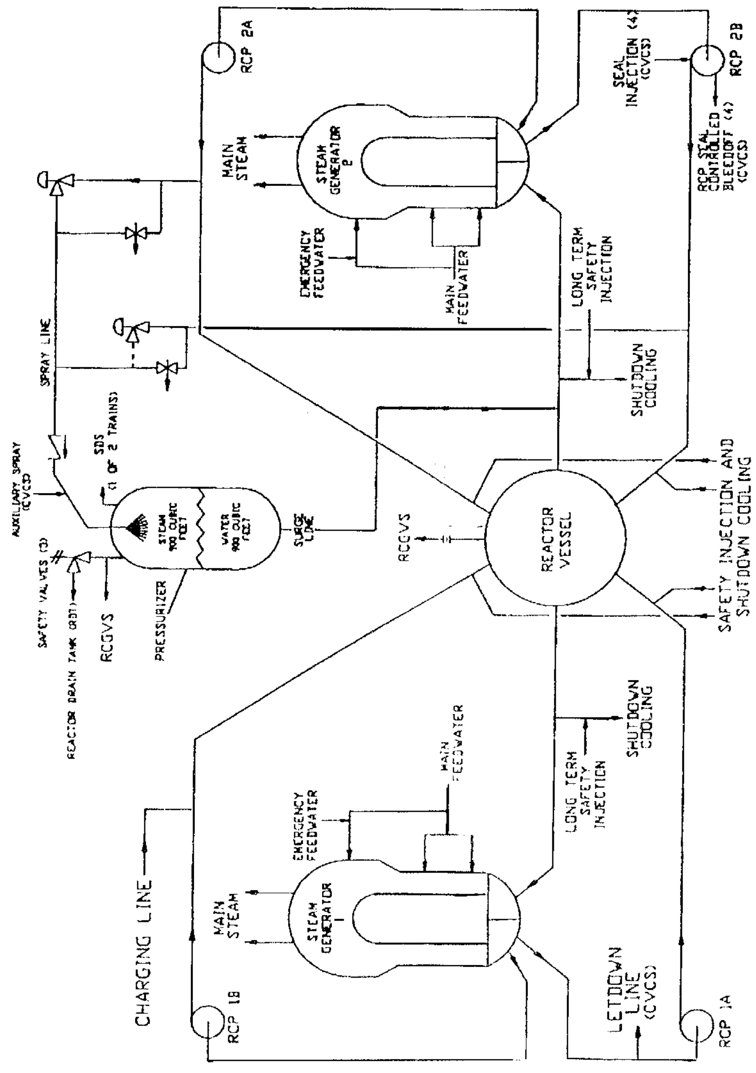


FIG. 4.8-1. KSNP$^+$- RCS Diagram

Overpressure protection of the reactor coolant system (RCS) is provided by two-stage depressurization through safety valves and safety depressurization valves (SDVs), which are mounted to the top of the pressurizer.

Overpressure protection for the shell side of the steam generators and the main steam-line piping, up to the inlet of the turbine stop valve, is provided by sixteen spring-loaded main steam safety valves (MSSVs). These valves are mounted on separate headers connected to the seismically designed portions of the main steam piping, equally divided between the main steam lines. The total relieving capacity of these valves is sufficient to pass excessive steam at 110% of the steam generator design pressure. The MSSV set pressure is calculated in accordance with Article NC-7000 of ASME Section III.

4.8.2.2 Reactor core and fuel design

The core of the KSNP$^+$ is designed to generate 2,825 MWt including 10 MWt of reactor coolant pump work. The reactor core is composed of 177 fuel assemblies and 73 control element assemblies (CEAs). The fuel assembly, which provides for 236 fuel rod positions (16 x 16 array), consists of five guide tubes welded to spacer grids, and is closed at the top and bottom by end fittings. In-core instrumentation is installed in the central guide tube of the fuel assembly. The in-core instrumentation is routed into the bottom of the fuel assembly through the bottom head of the reactor vessel. The CEAs consist of NiCrFe alloy-clad boron carbide absorber rods and solid NiCrFe alloy reduced-strength absorber rods, which are guided by tubes located within the fuel assembly.

The core is designed for an operating cycle of 18 months with a discharge burn-up of 58,000 MWd/t. A portion of the fuel rods contains uranium fuel mixed with a burnable absorber (Gadolinium) to suppress excess reactivity after fueling and to help control the power distribution in the core. The neutron flux shape is monitored by means of fixed in-core instrumentation (ICI) assemblies.

4.8.2.3 Fuel handling and storage system

The fuel handling system consists of the equipment and tools for refueling the reactor. The system is designed for safe and rapid handling and storage of fuel assemblies from receipt of new fuel to shipment of spent fuel.

The major equipment of the system comprises the refueling machine, the CEA change platform, the fuel transfer system, the new fuel elevator, the CEA elevator, and the spent fuel-handling machine. The refueling machine is located in the reactor containment building and moves fuel assemblies into and out of the reactor core and between the core and the fuel transfer system. The fuel transfer system rotates fuel assemblies from the vertical to the horizontal position, shuttles them through the fuel transfer tube assembly, and then returns them to the vertical position, either inside the reactor containment building or in the fuel handling building. The spent fuel handling machine, located in the fuel handling building, carries fuel to the shipping cask areas. The CEA change platform, which is located in the reactor containment building, is used to perform CEA replacement and is used as a work platform for handling and replacement of in-core instrumentation equipment. The CEA elevator, which is located in the reactor containment building, is used during CEA assembly replacement. The new fuel elevator, which is located in the fuel handling building, is used to lower new fuel into the spent fuel pool. A permanent pool seal assembly is utilized to seal the annulus between the reactor vessel flange and the refueling pool floor.

4.8.2.4 Primary components

Reactor Pressure Vessel

The reactor consists of a vertically-mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical closure head, internal structures, core support structures, fuel assemblies, control rod assemblies, and control and instrumentation facilities.

The major design improvements incorporated into the $KSNP^+$ reactor design include: enhancement of core monitoring capability, larger operating margins, higher power level, and lower failure rate of fuel elements for higher plant availability and reliability.

The reactor pressure vessel is manufactured by the ring-forging method to minimize welded parts. The inner surface of the reactor vessel is clad with austenitic stainless steel or NiCrFe alloy. The reactor vessel integrity is ensured for its lifetime against brittle fractures due to neutron irradiation embrittlement, by use of the well-characterized low alloy steel with low initial RT_{NDT}. In addition, it has been designed to have an end-of-life RT_{PTS} of 47.8°C (118°F).

Reactor Internals

The components of the reactor internals are divided into two major parts consisting of the core support-barrel assembly and the upper guide-structure assembly. The material used in fabrication of the reactor's internal structures is primarily austenitic stainless steel except for the holddown ring, which is made of martensitic stainless steel.

Steam generators

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. The heat transfer tubes are made of Inconel 690 to enhance the corrosion resistance, and have a plugging margin of 8% to improve the operating margin of the steam generator.

Pressurizer

The pressurizer, which is connected to one of the hot legs, consists of a steel pressure vessel containing pressurizer heaters, spray nozzles and safety valves. Its function is to maintain the pressure and water inventory of the reactor coolant system within specified limits during all normal and upset operating conditions without actuation of pressure relief devices.

Reactor coolant pumps

The reactor coolant pumps circulate reactor coolant through the reactor vessel to the steam generators for heat removal and then return it to the reactor vessel. There are two pumps for each coolant loop, located in each cold leg. The pump is a single-stage centrifugal unit of vertical type, driven by an 8800 hp electric motor. A mechanical seal designed to seal against the full internal pressure in the pump ensures leak-tightness of the shaft. The basic function and type of the pump in the $KSNP^+$ is the same as those of the KSNP.

Piping

The Leak-Before-Break (LBB) principle has been adopted for the piping system of the KSNP$^+$, since the pipe whip restraint and the support of the jet impingement shield in the piping system of the earlier plants 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, as well as the safety injection system in the containment.

4.8.2.5 Reactor auxiliary systems

Chemical and Volume Control System (CVCS)

Several design changes have been implemented in the CVCS of KSNP$^+$ for easier maintenance and more reliable operation. The capacities of large capacity tanks such as the Refueling Water Tank (RWT), Reactor Makeup Water Tank (RMWT), Holdup Tank (HT), and Equipment Drain Tank (EDT) have been optimized considering the level program of each tank and the margin for performance or safety. The seal injection subsystem has been changed to improve on the KSNP design by removing the Seal Injection Heat Exchanger (SIHX). Additionally, the safety and quality classes of the CVCS components have been adjusted based on the strict application of ASNI/ANS-51.1-1983 (reaffirmed 1988) and RG 1.26, respectively.

Reactor Coolant Gas Vent System (RCGVS)

Vent line connections to the RCGVS are provided from the inlet line of each pressurizer safety valve and the upstream of each Safety Depressurization System isolation valve. The vent line is used to vent air from the pressurizer prior to plant startup. In addition, the vent line allows non-condensable gases to be vented to the RCGVS during post-accident operations, when these gases may collect in the pressurizer steam space.

Steam Generator blowdown system

The functions of the SG blowdown system are to control SG secondary-side water chemistry and to remove sludge from the SG tube support plates. One flash tank can accommodate normal- and high-capacity blowdown flow rates. To remove dynamic loading due to two-phase flow, the flash tank for blowdown is located in the auxiliary building nearby the containment vessel.

Integrated Head Assembly (IHA)

The previous Reactor Vessel Closure Head (RVCH) area design for the KSNP had many components that needed to be disassembled, stored individually, and assembled during every refueling outage. In order to reduce the number of steps during the refueling process, beyond those of removing the studs and lifting of the head, the IHA concept has been introduced in the KSNP+ design.

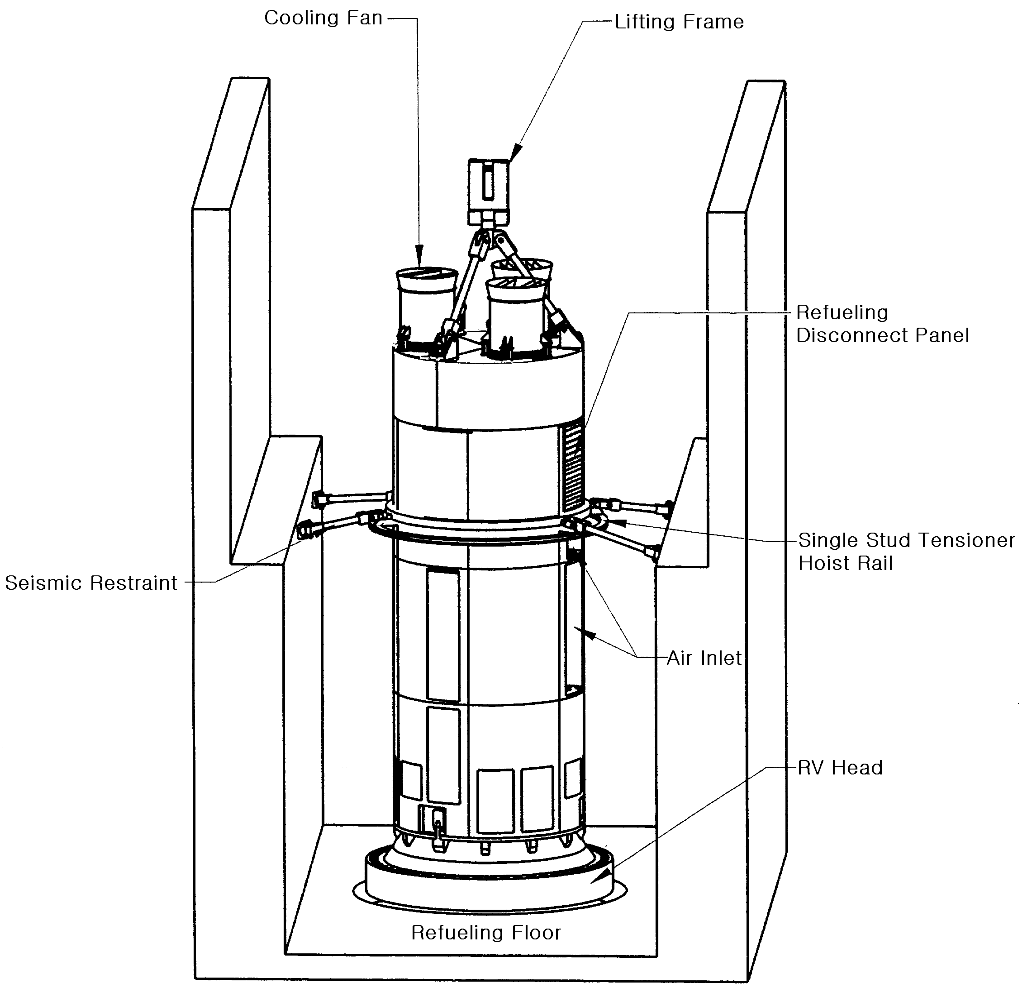


FIG. 4.8-2. KSNP$^+$ - integrated head assembly

The IHA, as shown in Fig.4.8-2, consists of the structurally-integrated equipment necessary for handling and storage of the RVCH and the additional equipment to perform functions for Control Element Drive Mechanism (CEDM) cooling, head-area cable support, missile shielding, and seismic loads transmitting from the CEDMs and the IHA to the refueling pool wall. During refueling, all these pieces of equipment are assembled together and moved to the storage stand as a single structure. The IHA will contribute to a reduction in radiation exposure to the head-area handling operators, as well as a reduction in the refueling outage duration.

4.8.2.6 Operating characteristics

The plant power control system is capable of daily load operations with a load variation profile typical in Korea: 16 hours at 100% and 4 hours at 50%, with 2 hours ramps for power decreases and increases. The reactor core control should be capable of a step power change of ±10% and ramp changes of 5% per minute without detrimental effects on the fuel rod integrity.

The load rejection capability at rated power should also be incorporated. Thus the power will not automatically be tripped in the event of a turbine trip if the transient causing the turbine trip is limited to the secondary system; assuming the primary system, including the reactor, is in normal operation condition. This capability can reduce the outage time caused by secondary system troubles, since the reactor power can be brought up to 100% as soon as the troubles have been fixed.

4.8.3 Description of turbine generator plant systems

4.8.3.1 Turbine generator plant

The turbine generator plant consists of main steam, extraction steam, feedwater, condensate, turbine generator and auxiliary systems.

The main turbine is of a tandem compound design consisting of a double-flow high-pressure turbine and three double-flow low-pressure turbines. The generator is a three-phase, 4-pole unit operating at 1,800 rpm.

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 to 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 to be operated for a steam pressure of 7.4 Mpa (1,070 psia).

4.8.3.2 Condensate and feedwater system

The condensate system condenses the low-pressure turbine exhaust steam, collects condensate in the hotwell, and pumps it to the de-aerator storage tanks. The condensate system consists of a single pressure condenser with three shells, three 50% capacity condensate pumps, condensate polishing demineralizers, three trains of three low-pressure feedwater heaters, a de-aerator, and two de-aerator storage tanks.

The feedwater system supplies feedwater from the de-aerator storage tank to two steam generators at the required pressure, temperature, and flow rate. Three motor-driven feedwater booster pumps and three turbine-driven feedwater pumps are provided. Each combination of a feedwater pump and a feedwater booster pump can provide a maximum of 55% of the flow requirements of the feedwater system.

4.8.3.3 Auxiliary systems

The auxiliary steam system is designed to provide process steam during plant startup, shutdown, and normal operation. It also provides steam for manual steam service in the reactor containment and fuel handling area.

The outdoor-type auxiliary boiler supplies steam to the auxiliary steam headers during plant shutdown and startup if steam is not available from other sources. The capacity of the auxiliary boiler is sufficient to meet steam requirements with one unit shut down and the other unit starting up.

The auxiliary boiler supplies steam to the system during shutdown or startup whenever steam is not available from the other sources. Main steam is supplied to the auxiliary steam system during plant normal operation. The auxiliary steam header is cross-tied for each unit.

4.8.4 Instrumentation and control systems

4.8.4.1 Design concept, including control room

The instrumentation and control systems and control room concept to be implemented in the KSNP+ design are schematically depicted in Figure 4.8-3.

Acknowledging the improved reliability of digital systems, the KSNP+ has been equipped with digitized instrumentation and control (I&C) systems and computer-based control room man-machine interface (MMI), along with conventional switches, reflecting the status of modern electronics and computer technologies.

The I&C and MMI of the KSNP+ are designed to have high reliability, safety, and maintainability, to perform plant control, monitoring, and protection functions efficiently during normal operation as well as in emergency situations, by implementing advanced digital technology along with a systematic application of human factor engineering principles.

The main features of the I&C system are the use of microprocessor based Programmable Logic Controller (PLCs) for the control and protection functions, and the use of UNIX workstations and computers for data processing systems. To protect against common-mode failures in the software due to wide use of software-based I&C systems, independence and diversity principles are applied. For safety systems, a thorough design verification and validation is performed to ensure high reliability. Different hardware platforms are selected for control systems for the sake of diversity. For data communication between systems, a high-speed fiber optic network is used, based on standard protocol. The I&C system architecture is designed, to every extent possible, with open technology based on industrial standards, for ease of upgrading and maintenance.

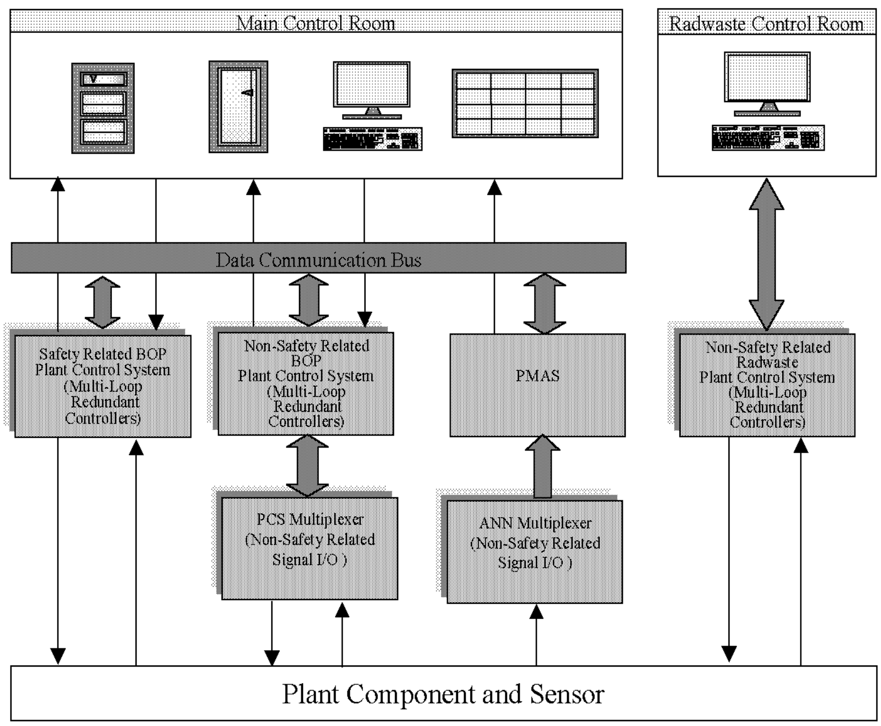


FIG. 4.8-3. KSNP+ - concept of I & C systems configuration

Since a lot of functions are implemented by means of software, verification and validation for software become very important. Systematic software verification and validation will be performed based on classification and function.

The KSNP$^+$ MCR contains the main control boards (MCBs), operator consoles, vertical panels, communication equipment, annunciators, CRTs, printers, and auxiliary equipment. The MCR is designed and arranged for the operators' error-free and safe operation of the plant as well as for operators' comfort, so as to reduce fatigue and stress.

The KSNP$^+$ radwaste control room design is characterized by a reduced number of dedicated conventional hardware-oriented indicators and annunciators. The radwaste control room is designed with a compact operator console and makes extensive use of visual display units such as LCDs (liquid crystal displays) and a large display unit for displaying radwaste system process information and operational status.

4.8.4.2 Reactor protection and other safety systems

The reactor protection system and other safety-related systems have been designed to meet the strict regulatory codes and standards that have been issued for digital safety systems. A high degree of reliability is required in safety-related systems, and therefore, design methodologies such as redundancy, diversity, and self-diagnostics have been incorporated in order to achieve both the desired reliability and availability of the systems.

The safety-related systems will be implemented using standard "off-the-shelf" hardware and software. The protection systems are designed with same hardware and software building blocks to reduce both development costs and maintenance costs during operation.

For the software development, an overall plan has been developed which defines the life cycle, the configuration management plan, and the verification and validation (V & V) plan. Based on its function, the software is divided into four classes. The documentation and V&V activities are different according to the classification of the software. For the safety software, safety analysis of the software and requirement traceability analysis is performed for each phase of software life cycle, and extensive software testing for each module and integrated application is performed.

For KSNP$^+$, the core protection calculator system (CPCS) has been upgraded with AC160 PLC. The AC160 platform has been used for the plant protection system (PPS) design in KSNP. Thus, AC160 hardware and software components are the building blocks for both protection systems: PPS and CPCS. To utilize standard off-the-shelf hardware and software, a commercial dedication process has been performed to show its adequacy in usage for safety application. During the upgrade, the multiple point-to-point Control Element Assembly(CEA)-position signal sharing using fiber optic modems between redundant channels has been replaced with signal sharing by data communication between redundant channels. The signal sharing by data communication allows reliable data sharing by failure diagnostics and makes the sharing simpler with one dedicated communication line for multiple signal sharing. The CEA Position Processor (CPP) is being implemented to perform CEA-position signal sharing using a dedicated processor board. The data communication to non-safety related systems is performed at the maintenance and test processor, which performs the role of communication buffer between safety and non-safety related systems.

4.8.4.3 Plant Monitoring and Annunciator System (PMAS)

The plant monitoring and annunciator system (PMAS) consists of the plant computer system (PCS), the plant data acquisition system (PDAS) and the plant annunciator system (PAS).

The PMAS incorporates distributed redundant system structures for high reliability, flexibility and expandability. Various operator support functions and applications have been developed and enhanced for the PMAS based on past experiences from plant operations and design evaluations. Those development and enhancements are as follows:

- Integration of plant monitoring system and plant annunciator system,
- Consistent and validated alarm information,
- Reduction of unnecessary redundant components and wiring,
- Sharing of peripherals between PMS and PAS,
- Incorporation of redundant distributed computer system architecture to enhance system reliability and availability,
- Improved plant computer system functions,
- SPADES (safety parameter display and evaluation system) for SPDS requirements,
- Real-time database with on-line modification capability,
- Historical data storage and retrieval function,
- Improved operator support functions,
- Fast data scan and SOE processing,
- On-line hardware and software diagnosis,
- Interface to plant LAN.

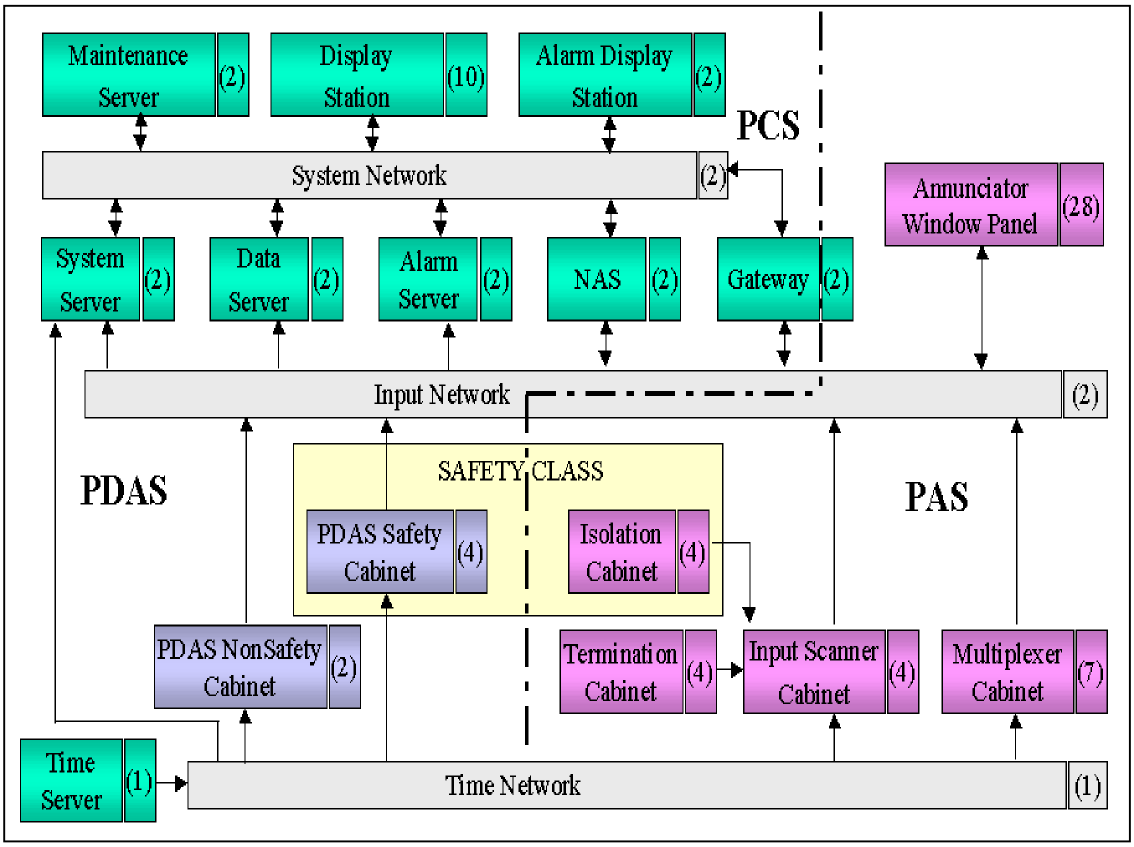


FIG. 4.8-4. KSNP+ - PMAS schematic diagram

4.8.4.4 Excore neutron flux monitoring system (ENFMS)

The ENFMS provides a means to measure the reactor power level by monitoring the neutron flux leakage from the reactor vessel for reactor control, protection and information display.

The ENFMS consists of four redundant safety channels and two redundant startup/control signal processing drawers. The safety channel provides the neutron flux information to be used by the reactor protection system. The startup/control signal processing drawer receives input signals from three safety channels through the qualified isolators and provides the source level neutron flux information and the power level neutron flux information. Each detector assembly, which consists of three fission chambers, is located in instrument wells around the reactor vessel cavity. A total of four instrument wells are required.

4.8.5 Electrical System

The main features of the electrical system configuration (Figure 4.8-5):

- Two independent offsite power sources: one of 154kV and the other of 765kV;
- One main transformer consisting of three single-phase step-up transformers, and two three-winding unit auxiliary transformers for power delivery and supply during normal operation mode;
- Two Class 1E emergency diesel generators to provide on-site standby power for the Class 1E loads;
- An alternate AC source to provide power for plant equipment necessary to cope with station blackout (SBO);
- Automatic transfer of power supply from unit auxiliary transformers to stand-by auxiliary transformers in the event of loss of power supply through the unit auxiliary transformers;
- Four independent Class 1E 125V DC systems for each reactor protection system channel of plant;
- Two Non-Class 1E 125V DC systems for each unit and two common Non-Class 1E 125V DC systems for the AAC and radwaste system in compound building;
- AC voltage levels of 13.8kV and 4.16kV for medium, 480V and 120V for low voltages.

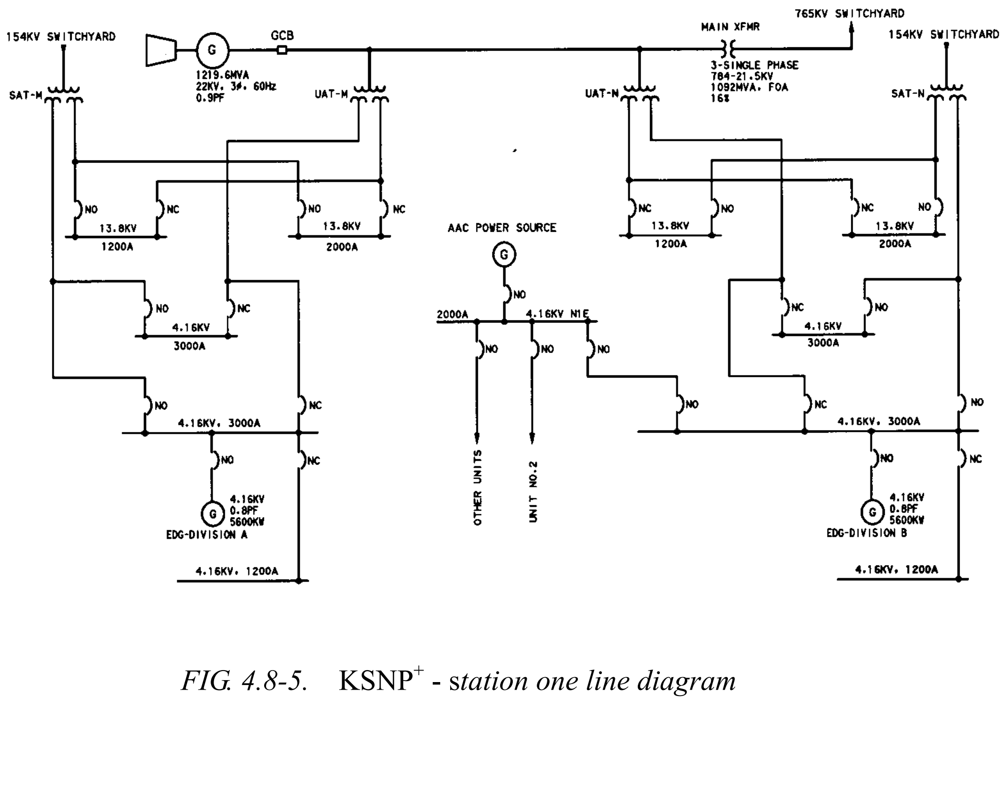


FIG. 4.8-5. KSNP+ - s*tation one line diagram*

4.8.5.1 Operational power supply systems

The main power system consists of the generator, the generator circuit breaker, the main transformer, the unit auxiliary transformers, and the standby transformers. The generator is connected to the Gas-Insulated 765kV switchyard via the main transformer, which is made up of three single-phase transformer units. The step-down unit auxiliary transformers are connected between the generator circuit breaker and the main transformer, and supply power to the unit equipment for plant startup, normal operation and shutdown. The stand-by transformer is always energized and ready to ensure rapid resumed power supply to the plant auxiliary equipment in the event of failure of the main and auxiliary transformers. The arrangement of the on-site electrical distribution system is based on the functional characteristics of the plant equipment to ensure reliability and redundancy of power sources.

4.8.5.2 Safety-related supply systems

The electrical power source feeding to the safety-related systems is supplied via four alternative means: first, the normal power source (i.e. the normal off-site power and the in-house generation); second, the standby power supply (i.e. two diesel generators); third, the onsite standby power supply (i.e. two diesel generators); and finally, the alternative AC source (i.e. the diesel generator that is installed to cope with station blackout).

Among these power sources, the on-site standby power is the most crucial for safety; it should be readily available in any situation. The arrangement of the on-site electrical distribution system is based on the functional characteristics of the equipment to ensure reliability and redundancy of all power sources.

The on-site power supply is ensured by two independent Class 1E diesel generator sets; each of them is located in a separate building and is connected to one 4.16kV safety bus.

The alternate AC source adds additional 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 a station blackout situation which has a high potential of transients progressing to severe accidents. The alternate AC source's size is of a sufficient capacity to accommodate the loads on safety which are required to achieve and maintain plant hot shutdown condition.

4.8.6 Safety concept

4.8.6.1 Safety requirements and design philosophy

The safety concept of KSNP$^+$ is based on the multiple level defense-in-depth approach: prevention of accidents or deviations from normal operation, detection of accidents through monitoring, control of accidents to prevent their propagation into severe accidents, and mitigation of severe accidents. The use of improved passive- and active-engineered safety features further reduces the probability of the occurrence of severe accidents.

This safety objective is pursued by compliance with deterministic requirements, supplemented by probabilistic methods. PSA techniques played an important role in the development of KSNP$^+$ design improvements. PSA was performed to find out the highest contributors and determine the rankings in system unavailability and safety impact among the candidate systems or items, prior to any design modifications. Also, to finally determine the integrated effects of design changes or modifications on plant safety, an on-going PSA approach has been adopted.

4.8.6.2 Safety systems and features

The safety systems consist of the safety injection system, safety depressurization system, shutdown cooling system, auxiliary feedwater system, and containment spray system.

Safety Injection System (SIS)

The SIS provides core cooling in the event of a loss-of-coolant accident (LOCA). The KSNP+ SIS consists of two 100% capacity trains. The SIS supplies sufficient cooling to remove the energy generated in the core for an extended period of time following a LOCA. The SIS also provides inventory and reactivity control during other events which depressurize the reactor coolant system, such as steam-line breaks and steam generator tube ruptures, and during feed-and-bleed operations. The SIS consists of two trains in two separate divisions. The principal components of the SIS are two high-pressure safety injection pumps, two low-pressure safety injection pumps, four safety injection tanks, and associated piping and valves.

Shutdown Cooling System (SCS)

The SCS is a forced-circulation heat removal loop designed to transfer heat from the RCS to the component cooling water system at temperatures where heat removal using steam generators is ineffective. The SCS consists of two separate divisions, each utilizing a low-pressure safety injection pump to circulate coolant through a shutdown-cooling heat exchanger. The RCS can be brought to refueling temperature using one shutdown cooling pump and one shutdown-cooling heat exchanger.

Containment Spray System (CSS)

The CSS is capable of reducing the containment pressure and temperature following a postulated loss-of-coolant accident (LOCA) or main steam-line break inside containment, removing radioactive fission products from the containment atmosphere, and mixing the containment atmosphere to prevent local accumulation of combustible gases following a postulated LOCA. The CSS consists of two separate trains each independently capable of meeting the ESF functional requirements. Each train includes a containment spray pump, a spray header, main and auxiliary spray rings, nozzles, valves, necessary piping, instrumentation, flushing connections, and related controls.

Auxiliary Feedwater System (AFS)

The AFS provides secondary-quality makeup water to the steam generators when the feedwater system is inoperable or unavailable. The AFS consists of two redundant trains. Each train consists of one 50% capacity motor-driven auxiliary feedwater pump, one 50% capacity turbine-driven auxiliary feedwater pump, associated valves, a cavitating venturi, piping, instrumentation, and related controls. The AFS operation is initiated by an auxiliary feedwater actuation signal due to low water level in the steam generator, or actuated manually from the main control board or the remote shutdown panel.

4.8.7 Plant layout

4.8.7.1 Buildings and structures, including plot plan

The general arrangement of the KSNP^{+} has been developed recognizing a need for improvement in the following aspects of the KSNP design:

- Improving the international marketability of the KSNP design through reduction of initial investment and operating costs;

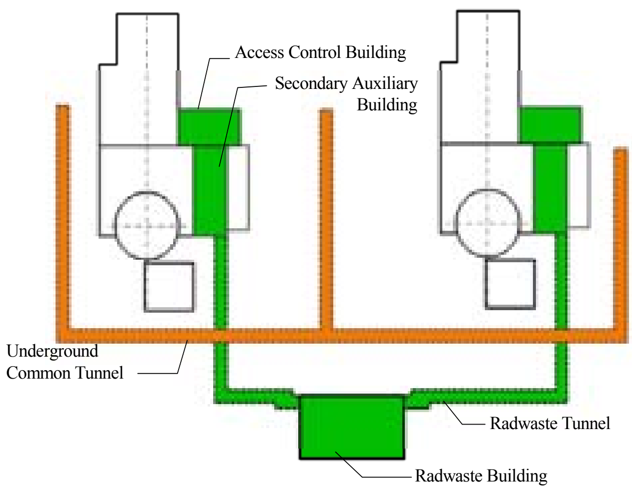
- Improving constructibility, operability and maintainability;
- Realizing self-reliance in nuclear power technology.

The plant arrangement of the KSNP^{+} is a “shared arrangement,” combining non-safety related buildings (five buildings of the two units) into one compound building, and a “slide-along arrangement” for all buildings except fuel handling and emergency diesel generator buildings.

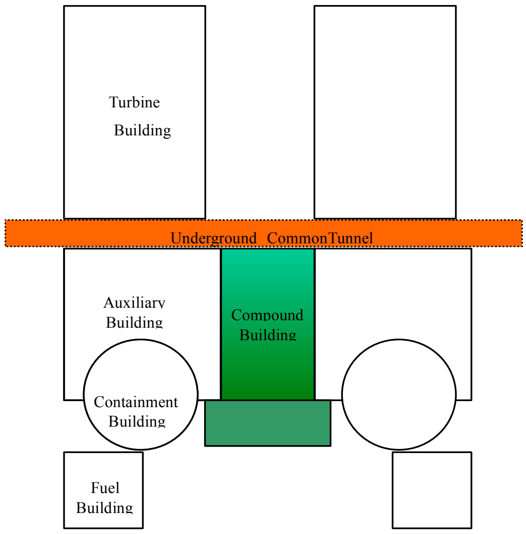
The main design features in plant arrangement are as follows.

- Combine non-safety related buildings into one compound building (Figure 4.8-6),
- Eliminate radwaste tunnel,
- Minimize the length of underground common tunnel,
- Minimize piping, cable tray, HVAC duct length,
- Arrange contaminated filter room and high level radwaste storage area closely above and below to Reduce occupational radiation exposure,
- locate hot machine shop in the power block.

This arrangement ensures significant improvements in constructibility, operability and maintainability, as well as reliability when compared to the original KSNP design.



< KSNP Plant Layout>



<KSNP^{+} Plant Layout>

FIG. 4.8-6 KSNP^{+} - general plant arrangement

4.8.7.2 Reactor containment building (RCB)

The reactor containment building consists of a steel-lined, post-tensioned concrete wall, and reinforced concrete internal structures. This RCB houses the reactor, steam generators, reactor coolant loops, refueling pool, and portions of the auxiliary systems.

The RCB 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 arrangement of the RCB is designed to meet the requirements for all anticipated conditions of operations and maintenance. There are four floor levels in the containment with the lowest being the basement at an elevation of 86 feet, and top floor being the operating floor at an elevation of 142 feet. The other two elevations are at 100 feet and 122 feet, with the one at 100 feet being the grade floor.

The containment is a post-tensioned concrete cylinder wall with a hemispherical dome. The cylindrical walls are 4 feet thick, and the dome thickness is 3 feet. The inside diameter of the containment is 144 feet, and it has an overall height of approximately 219 feet from the top of the basemat to the top of the dome.

The emergency exit airlock has been relocated to allow the installation of a temporary rail through the emergency exit airlock sleeve at the ground level for construction convenience**.** The equipment and structures in the two reactor containment buildings (such as the refueling canal, equipment hatch, and emergency exit air lock) are a mirror image of those in the fuel handling buildings.

4.8.7.3 Turbine generator building (TGB)

The turbine generator building houses the turbine generator, the condensers, the feedwater heaters, the condensate and feedwater systems, the main steam system, and other systems associated with power generation. The configuration of the TGB has been simplified for reasons of constructibility, and the maintainability of the systems has been improved by centralizing the condensate polishing system and the switch-gear room, and by rearranging the equipment hatches. There are four main floor levels referred to as the basement, the ground level, the operating level, and the de-aerator level.

4.8.7.4 Other buildings

Auxiliary building

The auxiliary buildings adjoins the reactor containment building and includes the main control room area, electrical and control area, and mechanical areas, which provide control and support functions to the reactor containment building. A major goal in the design of these areas was to create a safe and efficient environment for the people who work in the plant.

A major difference compared to the KSNP design is that the non-safety-related equipment contained in the secondary auxiliary building has been moved to the compound building. In addition to this, the flash tanks of the steam generator blow-down system have been moved to the auxiliary building from the turbine generator building to minimize vibrations problem due to transient loading. Also, the auxiliary building houses pumps and heat exchangers for the safety injection system and safety-related equipment required to provide safe shutdown capability.

Compound building

The compound building combines the functions of radwaste, access control, and those of the secondary auxiliary building of the KSNP, and minimizes the building volume by 73% compared with the total volume of KSNP's three buildings and tunnel configuration. The compound building houses the radwaste systems, chemical and volume control system, steam generator blow-down system, spent fuel pool cooling and cleanup system, radioactive laundry system, technical service centers, plant access control facilities and other miscellaneous systems for both units.

There are six floor levels referred to as the 2nd basement, the 1st basement, and 4 stories above ground level.

The compound building is classed as safety-related. It has no major structural interface with other buildings except for a seismic interface with the connecting auxiliary building. The compound building is located between the auxiliary building of each unit. The equipment, except that which is common to both units, are completely separated and located along the wall adjacent to each auxiliary building.

Fuel handling building

The fuel handling building is a reinforced concrete structure and consists primarily of a reinforced concrete pool. The fuel handling building contains the new and spent fuel storage pool, loading area, new fuel inspection area, new fuel storage pit, cask decontamination pit, fuel transfer canal, spent fuel ventilation system, refueling water tank, and the cask-loading pit. The cask pit is used when spent fuel is transferred from the spent fuel pool to the spent fuel-shipping cask.

4.8.8 Technical data

General plant data		
Power plant output, gross	1,050	MWe
Power plant output, net	1,000	MWe
Reactor thermal output	2,815	MWt
Power plant efficiency, net	35.4	%
Cooling water temperature (condenser)	28.5	℃
Nuclear steam supply system		
Number of coolant loops	2	
Primary circuit volume, including pressurizer	331.9	m^3
Steam flow rate at nominal conditions	1,603	kg/s
Feedwater flow rate at nominal conditions	1,606	kg/s
Steam Temperature/Pressure	289.4/7.38	℃/Mpa
Feedwater Temperature	232.2	℃
Reactor coolant system		
Primary coolant flow rate	16,593	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	295.8	℃
Coolant outlet temperature, at RPV outlet	327.3	℃
Mean temperature rise across core	31.5	℃
Reactor core		
Active core height	3.81	m
Equipment core diameter	3.12	m
Heat transfer surface in the core	4,842	m^2
Fuel inventory	76	t U
Average linear heat rate	17.69	kW/m
Average fuel power density	37.29	kW/kg U
Average core power density (volumetric)	96.4	kW/l
Total heat flux factor, Fq	2.52	
Rod radial power factor, Fxy	1.55	
Fuel material	Sintered UO_2	
Fuel assembly total length	4,527.6	mm
Rod array	square, 16x16	
Number of fuel assemblies	177	
Number of fuel rods/assembly	236	
Number of guide tubes	4 (per assembly)	
Number of spacers	11	
Enrichment (range) of first core	1.4/2.4/2.9/3.4	Wt. %
Enrichment of reload fuel at equilibrium core	4.6	Wt. %
Operating cycle length (fuel cycle length)	18	months
Average discharge burn-up of fuel	58,000	MWd/t
Cladding tube material	Zirlo	
Cladding tube wall thickness	0.635	mm
Outer diameter of fuel rods	9.7	mm
Overall weight of assembly	650.8	kg
Active length of fuel rods	3,810	mm
Burnable absorber, strategy/material	UO_2-Gd_2O_3	
Number of control rods	73	
Absorber rods per control assembly	4 or 12	
Absorber material (full/part. strength)	B_4C/Inconel 625	
Drive mechanism Positioning rate	40	steps/min
Soluble neutron absorber	Boron	
<u>Reactor pressure vessel</u>		
Cylindrical shell inner diameter	4,140.2	mm
Wall thickness of cylindrical shell	204.7	mm
Total height	14,642.3	mm
Base material: Cylindrical shell	SA508, Grade 3, Class 1	
RPV head	SA508, Grade 3, Class 1	
Liner	Stainless steel	
Design pressure/temperature	17.2/343.3	MPa/°C
Transport weight (lower part)	350.9	t (metric)
RPV head	67.8	t (metric)
Steam generators		
Number	2	
Heat transfer surface per SG	10,009	m^2
Number of heat exchanger tubes per SG	8,340	
Tube dimensions (outer diameter/thickness)	19.05/1.07	mm
Maximum outer diameter	4.412	m
Total height	22,987	mm

Transport weight	952.5	metric ton
Shell and tube sheet material	SA508 Grade 3	
Tube material	Inconel 690	
Reactor coolant pump		
Number	4	
Design pressure/temperature	17.2/343.3	MPa/℃
Design flow rate	7.67	m^3/s
Pump head	109.7	m
Power demand at coupling, cold/hot	9,933/7,409	kW
Pump casing material	SA 508, Class 2 or 3	
Pump speed	1,190	rpm
Pressurizer		
Total volume	50.9	m^3
Steam volume: full power/zero power	25.8	m^3
Design pressure/temperature	17.2/371.1	MPa/℃
Heating power of the heater rods	1,800	kW
Number of heater rods	36	
Inner diameter	2,438	mm
Total height	12,954	mm
Material	SA 533 Grade A/B, Class 1,	
SA 508, Class 2 or 3		
<u>Pressurizer Relief Tank (Reactor Drain Tank)</u>		
Quantity	1	
Internal Volume	20.8	m^3
Design Pressure (Internal)	896	KPa
Design Pressure (External)	276	KPa
Design Temperature	177	^{o} C
Rupture Disc (Diameter)	610	mm
(Internal Diameter	2,286	mm
Wetted Material	Austenitic stainless steal	
<u>Primary containment</u>		
Type	Pre-stressed and reinforced concrete	
Overall form (spherical/cyl.)	Cylindrical	
Dimensions (diameter/height)	43.9/65.8	m
Free volume	77,220	m^3

Design pressure/temperature (DBEs)		393.0/140.6	kPa/℃
Design leakage rate		< 0.1	vol.%/day
Is secondary containment provided?		no	
<u>Reactor auxiliary system</u>			
Reactor water cleanup,	capacity	9.5	kg/s
filter type	replaceable cartridge		
Residual heat removal,	at high pressure	N/A	kg/s
	at low pressure	315	kg/s
Coolant injection,	at high pressure	5.6	kg/s
	at low pressure	5.6	kg/s
<u>Power supply systems</u>			
Main transformer,	rated voltage	21.5/765	kV
	rated capacity	3x407.7	MVA
Plant transformers,	rated voltage	22/14.49, 4.47	kV
	rated capacity	63.9/71.6	MVA
Start-up transformer	rated voltage	154/13.8, 4.16	kV
	rated capacity	57.4/64.2	MVA
Medium voltage (6kV or 10kV)		13.8, 4.16	kV
Number of low voltage busbar systems		3	
Emergency Diesel generating units: number		4	
	rated power	5.6	MW
Number of diesel-backed busbar system		1 per DG unit	
Voltage level of these		4.16	kV ac
Number of DC distributions		4/2/3	
Voltage Level of these	125(Class 1E)/	250, 125(Non-1E)	V dc
Number of battery-backed busbar systems		4/1/2	
Voltage level of these		125(Class 1E)/	V ac
<u>Turbine plant</u>			
Number of turbines per reactor		1	
Type of turbine(s)	in-line, 6 flow, tandem regenerative Reheat TC6F-43		
Number of turbine section per unit (e.g. HP/LP/LP)		1 HP/3 LP	
Turbine speed		1,800	rpm

Overall length of turbine unit	60.5	m
Overall width of turbine unit	10.4	m
HP inlet pressure/temperature	6.8/287.1	MPa/ ^{o}C
<u>Generator</u>		
Type	4-pole, 1,800	rpm
Rated power	1,255	MVA
Active power	1,053.309	MW
Voltage	22	kV
Frequency	60	Hz
Total generator mass	2,674.0	t
Overall length of generator	10.6	m
<u>Condenser</u>		
Type	Once-through, sea water cooling	
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate	48,674	m^3/s
Cooling water temperature	28.5	^{o} C
Condenser pressure	38.1	mmHg
<u>Condenser pumps</u>		
Number	3 x 50 %	
Floe rate	544 (per pump)	kg/s
Pump head	247	m
Temperature	33/54.4	^{o} C
Pump speed	1,190	rpm
<u>Condensate clean-up system</u>		
Full flow/part. flow		
Filter type	Deep bed, mixed resin ion exchanger	
<u>Feedwater tank</u>		
Volume		m^3
Pressure/temperature		Mpa/ ^{o}C
Feedwater pumps		

Number	Turbine driven, 3 x 55 %	
Flow rate(per pump)	890	kg/s
Pump head	716.3	m
Feedwater temperature	137.5	^{o}C
Pump speed	5,500	rpm
<u>Condensate and feedwater heaters</u>		
Number of heating stages	7	
Redundancies	3 strings, 3 per string for LP feedwater heater	
	2 strings, 3 per string for HP feedwater heater	

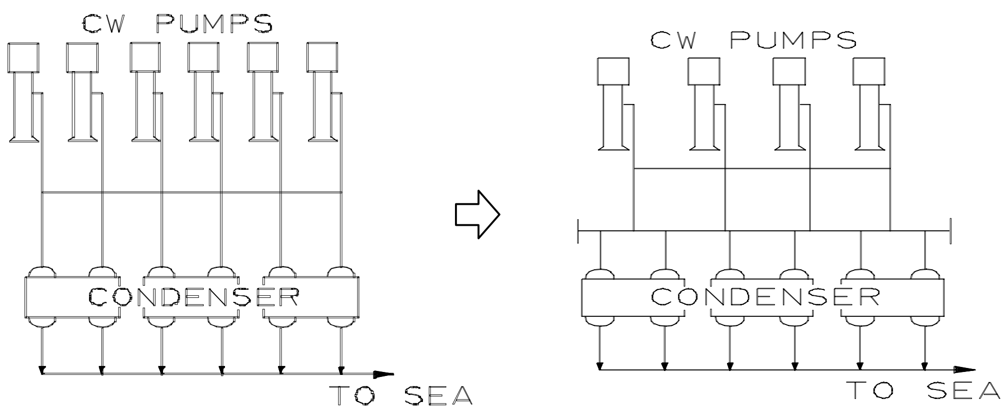
4.8.9 Measures to enhance economy and maintainability

Relative to the KSNP, the KSNP+ design measures taken to simplify the design, to reduce costs, construction schedule and the need for maintenance, to achieve high availability and flexibility of operation, and to improve the ability to perform maintenance are summarized as follows:

4.8.9.1 System Design Optimization

- Chemical and Volume Control System (CVCS)
  - Optimize the capacities of large CVCS Tanks and Letdown Heat Exchangers
  - Adjust safety and quality class by application of ANSI 51.1
  - Eliminate RCP Seal Injection Heat Exchangers;
- Plant Monitoring System (PMS) and Plant Annunciator System (PAS)
  - Unify PMS and PAS into Plant Monitoring & Annunciator System (PMAS) to eliminate redundant peripherals and human factor engineering inconsistencies;
- Circulating Water System (CWS)
  - Reduce the number of CWS Pumps and Travelling Screens from 6 to 4 per unit.

< KSNP Design> ***< KSNP+ Design>***



- Component Cooling Water System (CCWS)
  - Optimize system capacity by utilizing a common loop;
- Containment Spray System (CSS)
  - Eliminate dedicated heat exchanger(Containment Spray Heat-exchanger) by using the sharing shutdown cooling heat exchanger;
- Spent Fuel Pool Cleanup System (SFPCS)
  - Eliminate skimmer loop to cleanup SFP water surface and Reactor Cavity Filtration System(RCFS)
  - Connect the skimming suction to SFP cleanup system;
- Steam Generator Blowdown System
  - Combine two Continuous Blowdown (CBD) tanks and a High Capacity Blowdown (HCBD) tank into one
  - Eliminate non-regenerative heat exchanger and HCBD tank transfer pump.

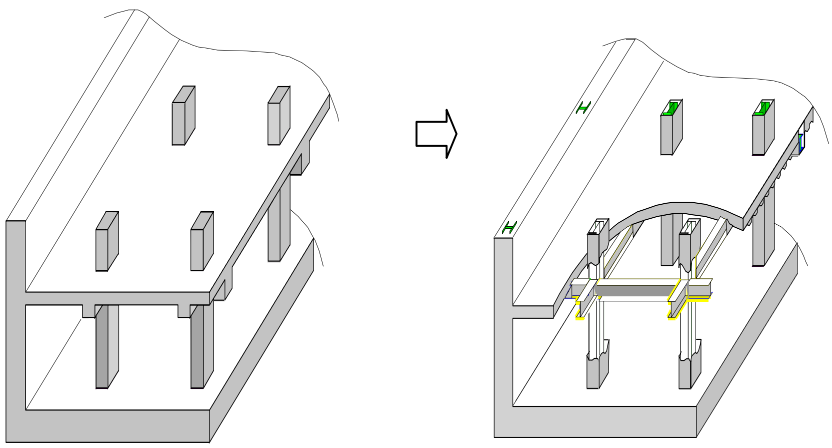
4.8.9.2 Optimization of equipment capacity and application of new technology

- Reduce capacity of Emergency Diesel Generator, Auxiliary Boiler, Condensate Pump, Stand-by Auxiliary Transformer;
- Replace the Active Hydrogen Recombiner of the Containment Hydrogen Control System with a Passive Autocatalytic Recombiner (PAR);
- Change filtering equipment for Liquid Radwaste System from Centrifuge Type to Reverse Osmosis Type;
- Change Control Method from Single Loop to Multi Loop of Plant Control System.

4.8.9.3 Application of steel-concrete composite structure

- Increase earthquake resistance and load bearing capacity
- Improve construction area availability by reducing structural member sizes
- Reduce construction resources
- Eliminate embedded plates on the ceiling
- Reduce the quantity of temporary construction structures

< KSNP Design> ***< KSNP$^+$ Design>***



4.8.9.4 Introduction of new construction methods to reduce construction period

- Apply area completion concept;
- Adopt deck plate construction method to apply to Steel-Concrete Composite Structure;
- Apply prefabrication and modularization for containment liner plate, rebar, support unit for deck plate, HVAC equipment and duct, and piping spool;
- Apply automatic welding process to reactor coolant loop piping;
- Apply jetty, access pit, and modularization in construction work to enhance constructibility.

4.8.9.5 Reduction of plant outage and replacement parts

- Adopt Integrated RV Head Assembly (IHA)
- Replace temporary type Refueling Pool Cavity Seal with permanent type
- Change the design of the Ex-core Neutron Flux Monitoring System with Long Life Fission Chamber type detectors
- Optimize the number and extend the service life of In-Core Instrumentations

4.8.10 Project status and planned schedule

The KSNP Design Improvement Program ($KSNP^+$ program) was launched at the beginning of 1998 and organized in three phases related to development status. Phase II of the program, which performed the basic design development, integration, verification, and licensing review for the design improvements selected during Phase I of the program, concluded at the end of 2001.

Phase III of the program is due to be completed in September 2008 for Unit 1 and September 2009 for Unit 2, with the completion of the construction of Shin-Kori Units 1&2 representing the first-fruits of the $KSNP^+$ program.

The major milestones of Shin-Kori 1 & 2 are as follows:

		# 1	#2
Excavation	:	Aug 1, 2003	Aug. 1, 2003
First Concrete		Feb. 1, 2004	Feb. 1. 2005
Set Reactor Vessel	:	Oct. 15, 2005	Oct. 15, 2006
Cold Hydro	:	May 1, 2007	May 1. 2008
Fuel Loading		Jan. 1, 2008	Jan. 1, 2009
Construction Completion		Sept. 30, 2008	Sept. 30, 2009

Bibliography

KHNP & KOPEC, ROK, Final Report of Korean Standard Nuclear Power Plant (KSNP) Design Improvement Phase I Project, Published Jan. 1999.

KHNP & KOPEC, ROK, Final Report of Korean Standard Nuclear Power Plant(KSNP) Design Improvement Phase II Project, Published Jan. 2001.

4.9 APR1400 (KOREA HYDRO AND NUCLEAR POWER COMPANY, REPUBLIC OF KOREA)

4.9.1 Introduction

The Advanced Power Reactor 1400 (APR1400), a standard evolutionary advanced light water reactor (ALWR) in the Republic of Korea, has been developed since 1992 with the name of Korean Next Generation Reactor (KNGR). The design is based on the experience that has been accumulated through the development of the Korean Standard Nuclear Power Plant (KSNPP) design, a 1000 MWe PWR. APR1400 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 the mitigation of severe accidents.

Since APR1400 is an evolution from its predecessor, the KSNPP, the basic configuration of the nuclear steam supply system is the same, i.e., it has two steam generators with four reactor coolant pumps in a "two hot legs and four cold legs" arrangement. However, the APR1400 has many advanced features such as the direct vessel injection of the safety injection system, in-containment refuelling water supply system, advanced safety depressurization system, and systems for severe accident mitigation. The power level of APR1400 is at least 1400MWe, which is 40% higher than that of the KSNPP. The main control room, designed with the consideration of human factors and digital I&C, is another example of the design improvement. Specially, the general arrangement has been improved with the reflection of operation and construction experiences of the KSNPP.

The reactor and plant design concept of APR1400 was based upon the results of two-year research in Phase I, which was finished in 1994. During this period, the Advanced Light Water Reactor (ALWR) designs being developed worldwide were reviewed and the design concepts were modified to meet the domestic needs and capabilities. The ALWRs developed worldwide were also reviewed quantitatively through safety and economic evaluations to establish the safety and economic goals for APR1400. The design requirements were established through this comparative study, and the major design requirements are in Table 4.9-I.

The basic design satisfying the above design requirements was developed during Phase II. Also, NSSS major component design specifications and Standard Safety Analysis Report (SSAR) were developed. At the end of basic design in early 1999, the review of design optimization was performed to improve the economic competitiveness, operability, and maintainability while maintaining the overall safety goal of the design. APR1400 finished Phase III as scheduled in 2001 and acquired the design certification from Korean regulatory agency in May 2002.

APR1400 was determined to be built as the next nuclear power plant in the Republic of Korea following 12 standard 1,000 MWe plants being operated or constructed. The site for APR1400 is decided near the Kori NPP site and the construction project for the twin units, Shin-Kori units 3&4, is in progress with the goal of commercial operation in 2010.

TABLE 4.9-I. APR1400 DESIGN REQUIREMENT FOR SAFETY AND PERFORMANCE GOALS

General Requirement	Performance requirements and economic goals
Type and capacity: PWR, 4000 MWt (NSSS system thermal power) Plant lifetime: 60 years Seismic design: SSE 0.3g Safety goals: Core damage frequency < 1.0E-5/RY Frequency of radiation release < 1.0E-6/RY Occup. radiation exposure < 1 man Sv per RY	Plant availability: greater than 90% Unplanned trips: less than 0.8 per year Refuelling interval: 18 months Construction period: 48 months (Nth plant) Economic goal: 20% cost advantages over competitive energy sources

4.9.2 Description of the nuclear systems

4.9.2.1 Primary circuit and its main characteristics

The primary loop configuration of APR1400 is similar to that of the KSNPP, which has two reactor coolant loops. The nuclear steam supply system is designed to operate at a rated thermal output of 4000 MWt to produce an electric power output of around 1450 MWe in the turbine/generator system. The major components of the primary circuit are the reactor vessel, two reactor coolant loops, each containing one hot leg, two cold legs, one steam generator (SG), and two reactor coolant pumps (RCPs), and a pressurizer (PZR) connected to one of the hot legs. The two steam generators (SGs) and the four RCPs are arranged symmetrically. The steam generators are located at a higher elevation than the reactor vessel for natural circulation purposes. For vent and drain, the elevation of the PZR and the surge line is higher than that of reactor coolant piping. A schematic diagram of arrangements and locations of the primary components and safety-related systems are shown in Figure 4.9-1.

In the reactor pressure vessel (RPV) design, four direct vessel injection (DVI) lines are connected to supply core cooling water from the in-containment refuelling water storage tank (IRWST). Level probes are added in the hot leg to monitor the water level during mid-loop operation. The design temperature in the hot leg is reduced from 327°C at the normal operating pressure, 15.5 MPa, of the currently operating nuclear plants to 324°C in order to reduce the possibility of Stress Corrosion Cracking (SCC) in S/G tubes. Conventional spring loaded safety valves mounted at the top of the PZR are replaced by the pilot operated safety relief valves (POSRVs), and functions of the RCS overpressure protection and rapid depressurization in case of severe accidents to prevent direct containment heating (DCH) shall be performed by the POSRVs. The POSRVs contribute to the safety enhancement due to higher reliability than the spring-loaded valves. On the secondary side of the SGs, two discharge trains are arranged on each main steam line at the outlet of the SG. Each train has five non-isolable safety valves, one main steam relief valve (normally closed), and one isolation valve (normally open).

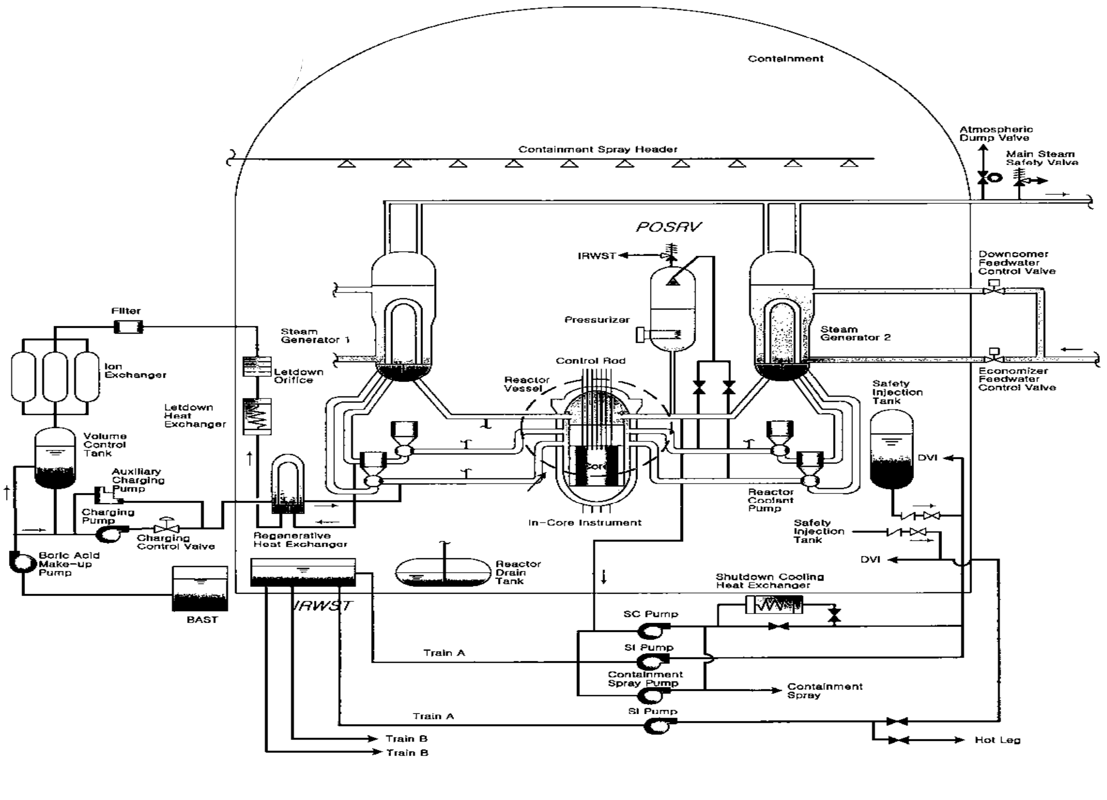


FIG. 4.9-1. Schematic diagram of primary components and safety systems

4.9.2.2 Reactor core and fuel design

The core of APR1400 is designed to generate an average volumetric power density of 250W/cm^3 of uranium fuel. The core consists of 241 fuel assemblies made of fuel rods containing uranium dioxide fuel with an average enrichment of 2.6 w/o in a 16×16 array. Each fuel assembly consists of 236 fuel rods and 5 guide tubes. The possibility of utilizing Mixed Oxide (MOX) fuel is considered in the core design, and additional Control Element Assemblies (CEAs) are installed to increase the reactivity control capability, if necessary, for MOX fuel loadings. The number of CEAs in the standard design is 93 with a reserve of 8 additional CEAs. 76 CEAs are full-strength reactivity control assemblies, and the rest are part-strength CEAs. The absorber materials used for full-strength control rods are boron carbide (B_4C) pellets. Inconel alloy 625 is used as the absorber material for the part-strength control rods.

The core is designed for an operating cycle of 18 months with a discharge burnup as high as approximately 60,000 MWD/MTU, and has an increased thermal margin of more than 10% to enhance safety and improve operation performance. A portion of the fuel rods contains uranium fuel mixed with a burnable absorber of gadolinium (GD_2O_3) to suppress excess reactivity after fuelling and to help control the power distribution in the core. The neutron flux shape is monitored by means of movable and fixed in-core instrumentation (ICI) assemblies.

4.9.2.3 Fuel handling and transfer systems

The fuel handling system is designed for safe and rapid handling and storage of fuel assemblies from the receipt of fresh fuel to the shipment of spent fuel.

The major equipment of the system comprises the refuelling machine, the CEA change platform, the fuel transfer system, the fresh fuel elevator, the CEA elevator and the spent fuel handling machine. The refuelling machine is located in the containment building and moves fuel assemblies into and out of the reactor core and between the core and the fuel transfer system. The spent fuel handling machine, located in the fuel building, carries fuel to and from the fuel transfer system, the fresh fuel elevator, the spent fuel storage racks and the spent fuel shipping cask areas.

The upper guide structure, which consists of the fuel assembly alignment plate, control element shroud tubes, the upper guide structure base plate, CEA shrouds, and an upper guide structure support barrel, is removed from the core as a single unit during refuelling by means of special lifting rig.

4.9.2.4 Primary components

Reactor pressure vessel

The reactor consists of a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical closure head, internal structures, core support structures, fuel assemblies, control rod assemblies, and control and instrumentation components.

The structural integrity of the reactor vessel is verified through the structural sizing and fatigue evaluation, which calculates the stresses of the heads, shell and nozzles under thermal and pressure loads.

The DVI nozzle is attached to the reactor vessel for the direct emergency coolant injection as a part of the safety injection system. The location of DVI nozzle is above the cold leg nozzles and determined to avoid the interference with reactor vessel external nozzles and support structure.

The life time of the reactor pressure vessel is extended to 60 years by the use of low carbon steel, which has lower contents of Cu, Ni, P, S compared with the current design, resulting in the increase of brittle fracture toughness. The inner surface of the reactor vessel is cladded with austenitic stainless steel or Ni-Cr-Fe alloy. The reactor vessel is designed to have an end-of-life RTndt of 21.1^{o}C(70^{o}F). Also, the reactor pressure vessel is manufactured by the ring forging method to minimize welding parts.

Reactor internals

The reactor internals consist of the core support structures, which include the core support barrel, upper guide structure barrel assembly and lower support structure, and the internal structures. The core support structures are designed to support and orient the reactor core fuel assemblies and control element assemblies, and to direct the reactor coolant to the core. The primary coolant flows in through the reactor vessel inlet nozzles from the reactor coolant pump, passes through the annulus between the reactor vessel and core support barrel, through the reactor vessel bottom plenum and core, and finally flows out through the outlet nozzles of the reactor vessel connected to the hot legs.

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 enhanced seismic requirements. All reactor internals are manufactured of austenitic stainless steel except for the hold-down ring, which is made of high-tension stainless steel. The hold-down ring absorbs vibrations caused by the load to the axial direction of internal structures.

Steam generators

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 in the steam generator design is the use of advanced corrosion resistant material in the steam generator tubes, i.e., Inconel 690 replacing Inconel 600. In order to improve the operating margin of the steam generator, the tube plugging margin increases from 8% in the earlier designs to 10% and the upper tube bundle system is optimized in order to minimize the tube thinning caused by secondary flow between tubes and their supports.

The feedwater nozzle provides a passage of feedwater up to the economizer installed to increase the thermal efficiency of the steam generator at the cold side, and experiences a high temperature gradient. The feedwater nozzles should endure the excessive thermal stress, which causes an excessively large fatigue usage factor. In order to meet the criteria of fatigue usage factor which is one of ASME code requirements, thermal sleeve inside the feedwater nozzle is installed.

The primary head of the SGs is designed with two pieces of forging since the size of the primary head is increased due to the increase of thermal power and tube plugging margin, while limiting the length of SGs as a result of the manufacturability-review of major heavy components.

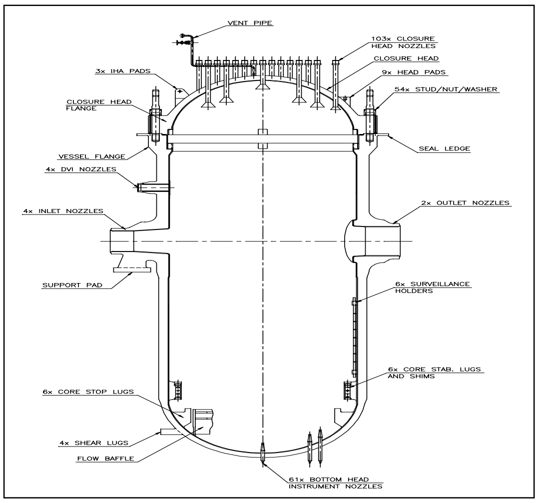


FIG. 4.9-2 *Reactor vessel layout*

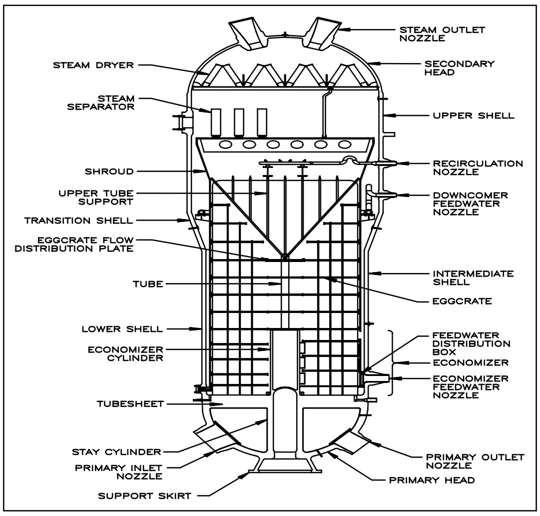


FIG. 4.9-3 Steam generator layout

Pressurizer

The pressurizer, having a total internal free volume of $68.9m^3$ (2,400 ft^3), is to maintain an operating pressure and temperature of the reactor coolant system. In the APR1400 pressurizer, the 3 pressurizer safety valves and the 2 safety depressurization system valves used for the KSNPP are replaced by the 4 Pilot Operated Safety Relief Valves (POSRVs). The POSRV has been verified through the installation and operating experience in the existing European and Canadian plants. It provides more reliability in overpressure protection function and more convenience in maintenance activities. The RCS inventory that would discharge through the POSRV under accident conditions is directed to the In-containment Refueling Water Storage Tank (IRWST) and quenched there so that the contamination of the containment environment is significantly reduced.

Integrated head assembly (IHA)

The IHA is a structure to combine and integrate all the reactor vessel closure head area structures into one assembly. The primary purpose of the IHA is to assemble all the head area structures, components, and cable system and their supports into one assembly so that the refueling time can be reduced from such operational activities as installation and removal of head area components. Also, the IHA contributes to the reduction of radiation exposures to the maintenance crew since the dissembling and assembling time of the reactor vessel head is reduced.

Reactor coolant pumps

The reactor coolant pumps circulate the coolant between the reactor vessel and the steam generators for heat transfer from the reactor core to the SGs. There are two pumps for each coolant loop, located in each cold leg. The pump is a single-stage centrifugal unit of vertical type, driven by a 13,320 hp electric motor. Leak-tightness of the shaft is ensured by a mechanical seal designed to prevent leaking against the full internal pressure in the pump.

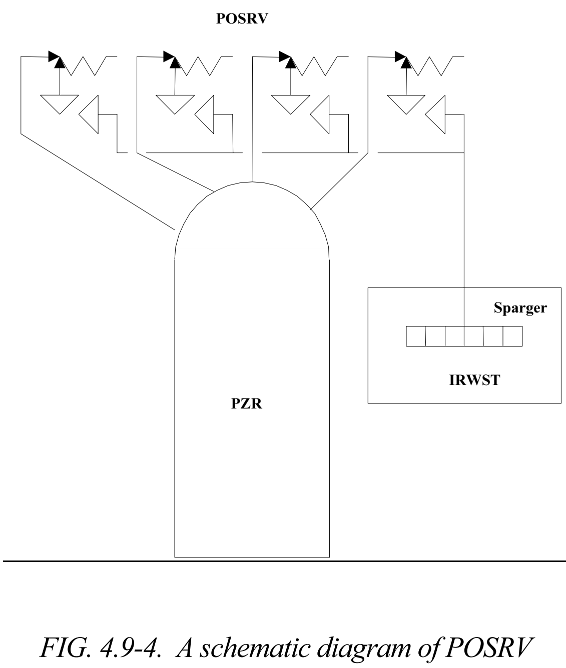


FIG. 4.9-4. A schematic diagram of POSRV

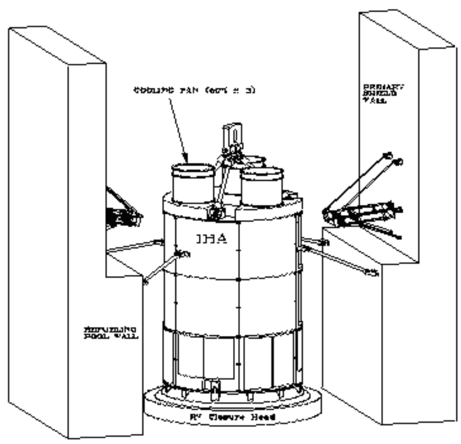


FIG. 4.9-5. A schematic diagram of IHA

Piping

The Leak-Before-Break (LBB) principle is adopted for the piping system of APR1400, since the pipe whip restraint and the support of the jet impingement shield in the piping system of earlier plants are expensive to build and maintain, and lead to a potential degradation of plant safety. The LBB principle is applied to the main coolant lines, surge lines, and pipes of the shutdown cooling system and the safety injection system. The application of LBB reduces the redundant supports of the pipe in the NSSS pipe system since the dynamic effects of postulated ruptures in the piping system can be eliminated from the design basis. Therefore, the cost of design, construction and maintenance is reduced.

4.9.2.5 Reactor auxiliary systems

Chemical and volume control system (CVCS)

The CVCS of APR1400 is not required to perform safety functions such as safe shutdown and accident mitigation. This system is basically for the normal day-to-day operation of the plant. The components related to charging and letdown function, however, are designed as a safety grade and reinforced to assure the reliability for normal and transient conditions. For normal operation, only one charging pump is used to supply the required minimum flow of 12.6 kg/s.

The letdown flow from the reactor coolant system passes through the regenerative and letdown heat exchanger, where an initial temperature reduction takes place. Pressure reduction occurs at the letdown orifice and the letdown control valve. Following temperature and pressure reduction, the flow passes through a purification process at the filters and ion exchangers. After passing through the purification process, the letdown flow is diverted into the volume control tank (VCT), which is designed to provide a reservoir of reactor coolant for the charging pumps and for the dedicated seal injection pumps for the reactor coolant pumps.

Component cooling water system

The component cooling water system (CCWS) is a closed loop cooling system that, in conjunction with the Essential Service Water System (ESWS) and Ultimate Heat Sink (UHS), removes heat generated from the plant's essential and non-essential components connected to the CCWS. Heat transferred by these components to the CCWS is rejected to the ESWS via the component cooling water heat exchangers. The system is designed to have the cross connection between two divisions to enhance the plant availability and maintenance flexibility.

Reactor coolant gas vent system (RCGVS)

The RCGVS is a part of the safety depressurization and vent system (SDVS). The reactor coolant gas vent valves are mounted at the top of the pressurizer. The size of the vent line is increased to have sufficient capacity to vent one-half of the RCS volume in one hour assuming a single failure. Moreover, flow-restricting orifices installed in the currently operating plants are removed from the system to improve vent capacity.

Steam generator blowdown system

The functions of the SG blowdown system are to control SG secondary side water chemistry and to remove sludge from the SG tube support plates. One flash tank can accommodate normal and high capacity blowdown flow rates. To remove dynamic loading due to two-phase flow, the flash tank for blowdown is located in the auxiliary building near the containment. Bypass lines to the condensers are installed to overcome unavailability of the flash tank or the processing system.

Primary sampling system

The primary sampling system is designed to collect and deliver representative samples of liquids and gases in various process systems to sample stations for chemical and radiological analysis. The system permits sampling during reactor operation, cooldown and post-accident modes without requiring access to the containment. Remote samples can be taken from the fluids in high radiation areas without requiring access to these areas.

4.9.3 Description of the turbine generator plant systems

4.9.3.1 Turbine generator plant

The turbine generator plant consists of the main steam, steam extraction, feedwater, condensate, turbine generator and auxiliary systems. For these systems, heat balance optimization was made considering system operability, reliability, availability and economy.

The turbine generator system is designed to be capable of operation at 3% house load for a period of at least 4 hours without any detrimental effects in the system, and capable of startup to full load from the cold condition 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 (1000 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 capacity, response and modulation capabilities of the turbine bypass system are designed to make the turbine capable of withstanding a 100% generator load rejection without trip of the reactor or the turbine. The total flow capacity of the turbine bypass system is designed to be 55% of the turbine steam flow at full load steam pressure.

4.9.3.2 Condensate and feedwater systems

The condensate and feedwater systems are designed to deliver the condensate water from the main condenser to the steam generator. The condensate pumps consist of three 50% capacity motor-driven pumps (two operating and one standby). The feedwater pump configuration is selected to be three 50% capacity turbine driven pumps because of its ability to allow more reliable operation; all three pumps are normally operating, and the plant can remain at 100% power operation even in the case that one of the feedwater pumps is lost.

During the shutdown and startup, a motor-driven startup feedwater pump provides feedwater from the deaerator storage tank or condensate tank. The startup feedwater pump is capable of providing up to 5% of full power feedwater flow to both steam generators. On-line condensate polishers, which 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 high reliability.

4.9.3.3 Auxiliary systems

Turbine bypass system

The turbine bypass system is provided to dissipate heat from the reactor coolant system during the turbine and/or the reactor trip. The KSNPP and the APR1400 plant have the same capability of relieving 55% of full load main steam flow. In the case of KSNPP, 15% are dumped to the atmosphere and 40% are discharged to the main condenser while the APR1400 plant discharges the total 55% directly into the main condenser.

Turbine building open cooling water system (TBOCW)

The TBOCW system supplies seawater to the service side of the turbine building closed cooling water (TBCCW) heat exchangers. The APR1400 plant does not need the TBOCW pump which is installed in the KSNPP to supply seawater as a heat sink for the plant. In the APR1400 plant design, the TBOCW system interfaces with the circulating water (CW) system to take the fresh seawater and discharge the heated seawater to the CW discharge conduit. This design concept reduces the plant capital cost.

Condenser vacuum system

The Condenser Vacuum (CV) system supports the plant startup and maintains the condenser vacuum by continuously removing non-condensible gases and air. The system consists of four 33-1/3 % capacity condenser vacuum pumps which are used to draw down the condenser shell pressure. These pumps are also used for "Holding mode" during normal operation without the steam jet air ejectors. In addition, the radiation level in the CV system discharge is continuously displayed on the radiation monitoring system in the main control room. The APR1400 plant is designed to combine the system discharge and the deaerator normal vent flow line to reduce the number of radiation monitors.

4.9.4 Instrumentation and control systems

4.9.4.1 Design concept including control room

APR1400 is, like most of the advanced reactors being developed world-wide, equipped with digitized instrumentation and control (I&C) systems and computer-based control room man-machine interface (MMI), reflecting the status of modern electronics and computer technologies. The I&C and control room concept implemented in the APR1400 design is schematically depicted in Figure 4.9-6.

The APR1400 I&C system is designed with the network-based distributed control architecture. In this architecture, operator interface functions and control functions for NSSS, BOP and TG are integrated in common design standards and implemented in common digital system for high functionality, easy operation, and cost effective maintenance. Diversity between safety I&C systems and non-safety I&C systems together with hardwired switches are provided for the defense-in-depth against common mode failure of software in the safety I&C systems.

The main features of the I&C system are the use of distributed control system (DCS) and microprocessor-based Programmable Logic Controllers (PLCs) for the control and protection systems, and the use of UNIX workstations and industrial PCs (personal computers) for data processing systems.

To protect against common mode failures in software due to the use of software-based I&C systems, DCS and PLCs will be required in the redundant systems for diversity. For data communication, a high-speed fibre optic network based on standard protocols is used. The remote signal multiplexer is also utilized for the safety and non-safety systems field signal transmission.

Human factor engineering is an essential element of the MCR design and the human factor engineering principles are systematically employed to ensure safe and error-free operation. For the successful completion of the APR1400 MMI design process, a multidisciplinary team of human factor specialists, computer specialists, system engineers, and plant operators worked together as a team from the stage of conceptual design through the validation process.

4.9.4.2 Reactor protection and other safety systems

The plant protection system (PPS) includes the electrical, electronic, networking, and mechanical devices to perform the protective functions via the reactor protection system (RPS) and engineered safety features actuation system(ESFAS). The RPS is the portion of the PPS that acts to trip the reactor when the monitored conditions approach specified safety settings and the ESFAS activates the engineered safety systems by safety injection actuation signal and the auxiliary feedwater actuation signal, and etc.

The reactor protection system and other safety-related systems are designed to use the off-the-shelf digital equipment which is commercially available to standardize the components and minimize the maintenance cost with the consideration of diversity. A high degree of conservatism is required in the design of the safety-related systems, and therefore, design principles such as redundancy, diversity, and segmentation have been incorporated in order to achieve both the desired availability and reliability of these systems.

A high reliability of the protection system is ensured by self-diagnostics, and automatic functional tests through surveillance using four independent channels. The redundant and fault tolerant configuration on controllers and the use of fiber-optics to isolate communications will increase system availability and maintainability.

A detailed software development program for software-based Class 1E systems were produced and applied as a guideline to ensure completeness of the software implementation, verification and validation process. Several critical safety systems were evaluated through prototyping and design verification programmes.

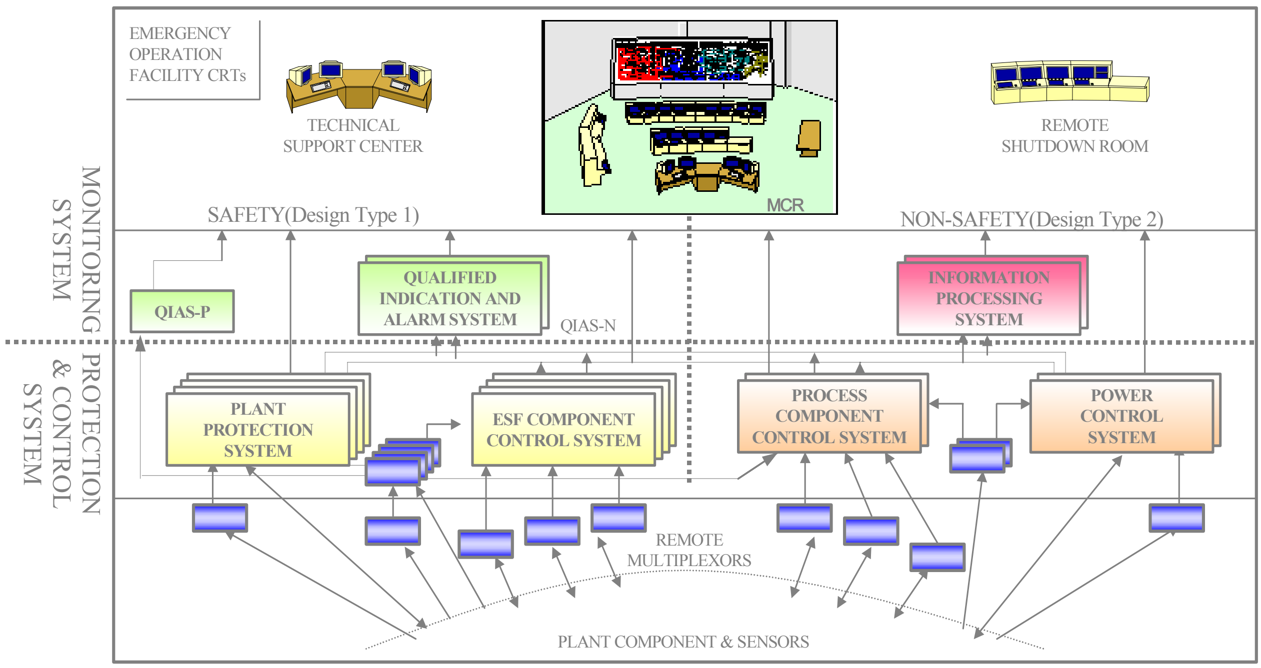


FIG. 4.9-6. Concept of MMIS configuration

4.9.5 Electrical systems

The one line diagram of APR1400 is shown in Figure 4.9-7. The main features of the electrical system configuration are:

- Two independent off-site power sources of 345 kV;
- One main transformer consisting of three single-phase step-down transformers, and two three-winding unit auxiliary transformers for power delivery and supply during normal operation mode;
- Two Class 1E emergency DGs to provide on-site stand-by power for the Class 1E loads;
- An alternate AC source to provide power for equipment necessary to cope with station blackout at least for 8 hours. For the diversity of emergency electrical power sources, the gas turbine type is selected for AAC;
- Automatic transfer of power source from unit auxiliary transformers to standby auxiliary transformers in the event of loss of power supply through the unit auxiliary transformers;
- Four independent Class 1E 125V DC systems for each RPS channel;
- Two non-class 1E 125V DC systems and one non-class 1E 250V DC system ;
- AC voltage levels of 13.8 kV and 4.16kV for medium, 480 V and 120V for low voltages.

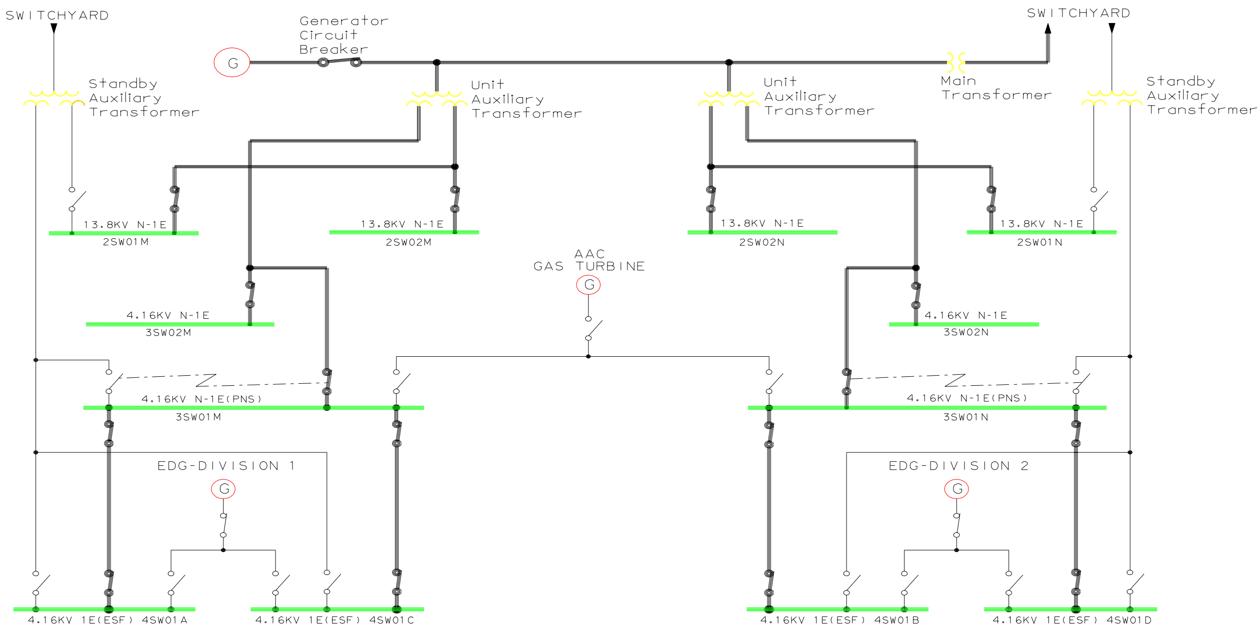


FIG. 4.9-7. APR1400 one line diagram

4.9.5.1 Operational power supply systems

The main power system consists of the generator, generator circuit breaker, main transformer, unit auxiliary transformer and stand-by transformer. The generator is connected to a gas-insulated 345 kV switchyard via the main transformer which is made of three single-phase transformer units. Step-down unit auxiliary transformers are connected between the generator and main transformer, and supply power to the unit equipment for plant startup, normal operation and shutdown. The stand-by transformer is always energized and ready to ensure rapid power supply to the plant auxiliary equipment in the event of failure of the main and unit auxiliary transformers.

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.

4.9.5.2 Safety-related systems

The electric power necessary for the safety-related systems is supplied through 4 alternative ways: firstly, the normal power source, i.e., the normal off-site power and the in-house generation; secondly, the stand-by off-site power, i.e., the off-site power connected through the stand-by transformer; thirdly, the on-site standby power supply, i.e., two diesel generators; and finally, the alternative AC source, i.e., the gas turbine generator.

Among these power sources, the on-site standby power is the most crucial for safety; it should be available in any situation. The arrangement of the on-site electrical distribution system is based on the functional characteristics of the equipment to ensure reliability and redundancy of power sources.

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 Loss-of-Off-site-Power (LOOP) and Station Blackout (SBO) situation which have a high potential of transients leading 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.

4.9.6 Safety concept

4.9.6.1 Safety requirements and design philosophy

Safety is a requirement of paramount importance for nuclear power. One of the APR1400 development policies is to increase the level of safety significantly. Safety and economics in nuclear power plants are not counteracting each other but can move in the same direction, since the enhancement of safety will also yield an improved protection of the owner's investment. Therefore, safety has been given top priority in developing the new design. To implement this policy, in addition to the plant being designed in accordance with the established licensing design basis to meet the licensing rules, APR1400 was designed with an additional safety margin in order to improve the protection of the investment, as well as the protection of the public health.

In order to implement this safety objective, quantitative safety goals for the design were established in a probabilistic approach:

- The total core damage frequency should not exceed 10E-5 per year, considering both internal and external initiating events. In addition, the frequency of core damage with reactor coolant pressure remaining high should not exceed 10E-6 per year.

- The whole body dose for a person at the site boundary should not exceed 0.01 Sv (1 rem) during 24 hours after the initiation of core damage, even in the event of containment failure. The probability frequency of exceeding such a limit should be less than 10E-6 per year.

To achieve the above quantitative goals, the defense-in-depth concept remains as a fundamental principle of safety, requiring a balance between accident prevention and mitigation. With respect to accident prevention, the increased design margin and system simplification represent a major design improvement and the consideration of accident mitigation call for the incorporation of design features to cope with severe accidents as well as design basis accidents.

In addition to the public safety, a concept of investment protection is implemented in APR1400 design. An example of a design requirement that aims at investment protection is the stipulation that a small break loss-of-coolant-accident (LOCA) with a break size smaller than 150 mm in diameter should allow the continued use of the reactor with its fuel inventory after the repair of the ruptured pipe (and/or other damages in the reactor coolant system).

The enhanced margin could benefit the operability and availability of the nuclear power plants. For example, the margin can alleviate transients, thereby avoiding unexpected trips, and be used for later system modification or adaptation of new regulatory restriction. A few examples of the design requirements following this philosophy are the requested core thermal margin of more than 10%, sufficient system capacity for the prolonged operator response time on the transient events, and station blackout coping time.

4.9.6.2 Safety systems and features (active, passive and inherent)

The major safety systems are the safety injection system, safety depressurization and vent system, in-containment refuelling water storage system, shutdown cooling system, auxiliary feedwater supply system, and containment spray system. A schematic diagram of arrangements and locations of safety systems is shown in Figure 4.9-1.

Safety injection system (SIS)

The main design concept of the Safety Injection System (SIS) is simplification and diversity 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. To satisfy the LOCA performance requirements, for breaks larger than the size of an direct vessel injection line (i.e., 216 mm), each train provides 50% of the minimum injection flow rate and, for breaks equal to or smaller than the size of an injection line, each train has 100% of the required capacity. The common header currently installed in the KSNPP SIS trains is eliminated and the functions for safety injection and shutdown cooling are separated.

Through the In-containment Refuelling Water Storage Tank (IRWST) system, the current operation modes of high pressure, low pressure, and re-circulation are merged into only one operation mode in case of LOCAs. Accordingly, the low pressure pumps are eliminated from the SIS and water source for the safety injection is taken from the IRWST only. The core cooling water is designed to be injected directly into the reactor vessel so that the possibility of the spill of the injected flow through the broken cold leg is eliminated. For this purpose, four DVI lines are connected with the reactor vessel. The DVI lines are installed between the hot and cold legs and are located above the locations of the hot and cold legs.

In-containment refuelling water storage system(IRWSS)

The refuelling water storage tank (IRWST) is located inside the containment and the arrangement is made in such a way that the injected core cooling water can return to the IRWST. It consists of an annular cylindrical tank surrounding the inside of the containment, holdup volume tank (HVT), and reactor cavity. The susceptibility of the current refuelling water storage tank to external hazard is lowered by locating it inside the containment. The IRWST provides the functions of storage for refuelling water, a single source of water for the safety injection, shutdown cooling, and containment spray pumps.

The IRWST is also used as a heat sink to condense the steam discharged from the pressurizer in case of rapid depressurization of the RCS to prevent high pressure core melt or to enable feed and bleed operation. Moreover, it provides the function of coolant supply to the cavity flooding system in case of severe accidents to protect against the molten core. The volume of the IRWST is 2470 m^3 (652 800 gal). This capacity is sufficient for flooding the refuelling cavity during normal operations, assuming the initial RCS level is at the center line of the hot leg. It also covers the capacity (i.e., 833 m^3) to flood the HVT and the reactor cavity to mitigate the impact of severe accidents.

Shutdown cooling system (SCS)

The shutdown cooling system is a safety-related system that is used in conjunction with the main steam and main or auxiliary feedwater system to reduce the temperature of the RCS in post shutdown periods from the hot shutdown operating temperature to the refueling temperature. After initial heat rejection from the SGs to the condenser or atmosphere, the SCS is put into operation at 176.7 ^{o}C and 31.6 kg/cm^2 A.

To improve shutdown cooling capability and system reliability, and to remove any possibility of intersystem LOCA, the following improvements were implemented for the design of the SCS:

- Increase of the design pressure to 6.2 MPa to protect the inter-system LOCA;
- Reinforcement of decay heat removal function for the reactor emergency conditions;
- Adoption of the partial 4 train concept by introducing exchangeable shutdown cooling and containment spray pumps; and
- Installation of the independent heat exchanger.

Auxiliary feedwater system

The Auxiliary Feedwater System (AFWS) is a dedicated safety system designed to supply feedwater to the SGs for removal of heat from the RCS for events in which the main or startup feedwater systems are unavailable. In addition, the AFWS refills the SGs following the steam generator tube rupture to minimize leakage through the ruptured tubes. The auxiliary feedwater system is a 2-division and 4-train system like the SIS. The reliability of the AFWS has been increased by use of two 100% motor-driven pumps, two 100% turbine-driven pumps and two dedicated safety-related auxiliary feedwater storage tanks as a water source in addition to the non-safety grade condensate storage tank.

Fluidic device

A passive flow regulating device, named fluidic device, is installed in each safety injection tank (SIT) to provide two operation stages of cooling water injection into the RCS. It is a passive system to inject the borated water into the RCS in a passive way with a capability of reducing the discharge flow at certain point to 20% of the maximum flow. This device makes the use of the SIT inventory more efficient since the SIT inventory can be used for a prolonged period.

Experimental test for validation of new design features

APR1400 adopted new safety features such as the safety injection system with direct vessel injection, POSRV with spargers, and fluidic device. To verify these new features, associated test programs were conducted. Also, the fatigue test of control element drive mechanism was carried out in order to verify the extended design life time.

4.9.6.3 Severe accidents (beyond design basis accidents)

The most advanced feature for safety from the current nuclear power plant design may be the inclusion of severe accidents mitigation in the design. All ALWRs currently developed have design features addressing the severe accident issues in one way or another. In APR1400, severe accidents are addressed as follows:

- For phenomena likely to cause early containment failure, for instance, within 24 hours after accidents, mitigation systems shall be provided or design should address the phenomena although the probability for such accidents is low.

- For phenomena which potentially lead to late containment failure if not properly mitigated, the mitigation system or design measures should be considered in conjunction with the probabilistic safety goal and cost for incorporating such features to address the phenomena.

This approach is to enhance the effectiveness of investment on safety by avoiding undue over-investment on highly improbable accidents. Also, a realistic assessment is recommended for severe accident analyses. More specific design features for mitigation of severe accidents are as follows:

Containment Hydrogen Control System (CHCS)

The CHCS is composed of three subsystems: the passive hydrogen recombiner system (PHRS), the containment hydrogen purge system(CHPS), and the hydrogen mitigation system(HMS). Following a LOCA, control of combustible gas concentration in the containment is provided by the PHRS. The PHRS prevents the concentration of hydrogen from reaching the lower flammability limit of 4% by volume in air or steam-air mixtures. In addition, the CHPS provides the capability for controlled purging to reduce the hydrogen concentration in the post-accident containment atmosphere with filtration of the discharge through the containment purge system.

During degraded core accident, hydrogen will be generated at a greater rate than during the design basis LOCA. The HMS is designed to accommodate the hydrogen production from 100% fuel clad metal-water reaction and limit the average hydrogen concentration in containment to 10% for degraded core accidents. The HMS consists of a system of Passive Auto-catalytic Recombiners (PARs) complemented by glow plug igniters installed within the containment. The PARs serve for accident sequences in which mild or slow release rates of hydrogen are expected, and are installed uniformly in the containment. Whereas, the igniters supplement PARs under the accident of very low probability in which very rapid release rates of hydrogen are expected, and are placed near source locations to facilitate the combustion of hydrogen in a controlled manner such that containment integrity is maintained.

External Reactor Vessel Cooling System (ERVCS)

The ERVCS is implemented as a severe accident mitigation system used for the purpose of in-vessel retention of corium under hypothetical core-melting severe accident conditions. The ERVCS shall be used only under the severe accident condition and thus is designed on safety margin basis. As shown in Fig. 4.9-8, one train of shutdown cooling pump, with related valves, pipes, and instrumentation & controls, is provided for initial reactor cavity flooding to the level of hot leg. After the initial flooding by the SCP, the boric acid makeup pump (BAMP) is utilized to refill the reactor cavity, at a flow rate greater than that of boiling caused by decay heat from the molten core. The ERVCS is designed to be manually operable only when the core exit temperature reaches a certain temperature following a severe accident. The operating procedure for the ERVCS was developed through severe accident analysis and probabilistic safety assessment. The gravity driven cavity flooding system (CFS) provides flooding of the reactor cavity below the reactor vessel. The CFS is a backup system used in case that ERVCS is unavailable and provides corium cooling, should the reactor vessel fail.

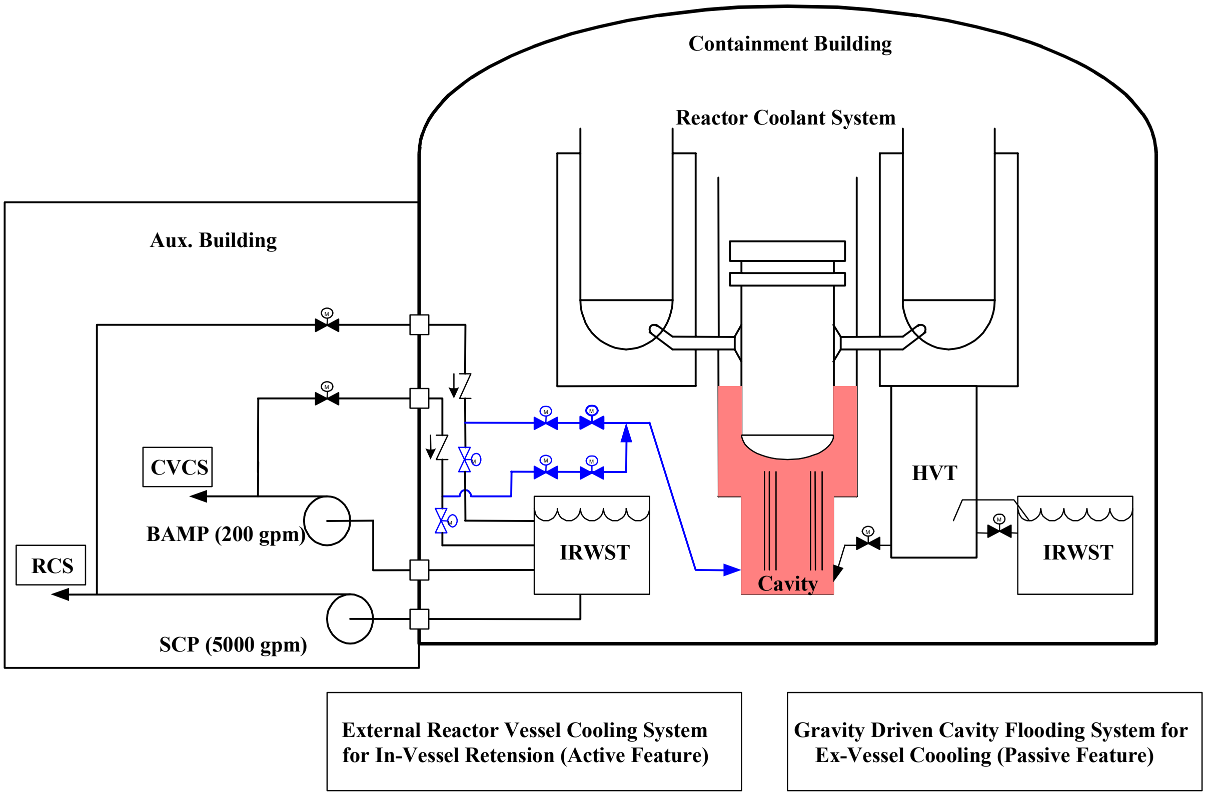


FIG. 4.9-8 External reactor vessel cooling system and gravity driven cavity flooding system

4.9.7 Plant layout

4.9.7.1 Buildings and structures, including plot plan

The general arrangement of APR1400 was designed based on the twin-unit concept and slide-along arrangement with common facilities such as the compound building which includes radwaste building and access control building. The general arrangement of the buildings is schematically depicted in Figure 4.9-9. The auxiliary building which accommodates the safety systems and components surrounds the containment building. 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 layout is highly influenced by safety considerations, in particular, by the physical separation of equipment for the safety systems. The safety injection pumps are located in the auxiliary building in the four quadrants, one pump in each quadrant. This arrangement ensures the physical separation of the pumps, minimizing the propagation of damage due to fire, sabotage, and internal flooding. The emergency diesel generator rooms are also separated and located at the symmetrically opposite sides.

The building arrangement is also designed for the convenience of maintenance, considering accessibility and replaceability of equipment. The internal layout of the containment, in particular, is designed to allow the one-piece removal of the steam generator. With proper shielding and arrangement of maintenance space, and careful routing of ventilation air flow, the occupational radiation exposure is expected to be lower than 1 man-Sievert a year.

The design strength of the buildings in the safety category, which are the containment and the auxiliary buildings, is sufficient to withstand the effects of earthquakes up to the safe shutdown earthquake (SSE) of 0.3 g.

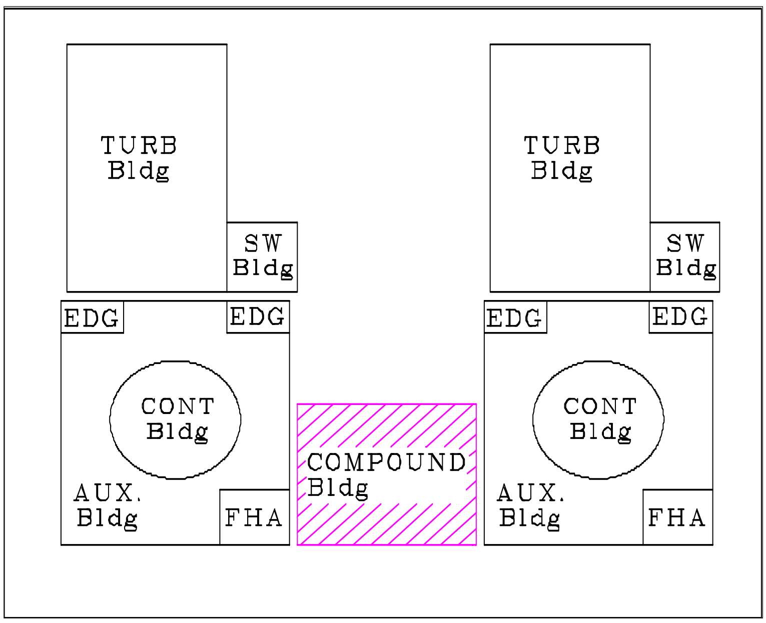


FIG. 4.9-9. Plant general arrangement

4.9.7.2 *Reactor building*

The reactor building is the central building of the plant. APR1400 is a pressurized water reactor, and the reactor building essentially coincides with the containment building. Figure 4.9-10 shows a cross-sectional view of the reactor building including a part of the auxiliary building in the vertical direction with the arrangement of major equipment.

4.9.7.3 *Containment*

The containment building is made of the post-tensioned cylindrical concrete wall with a steel liner, and reinforced concrete internal structures. The containment building houses a reactor, steam generators, reactor coolant loops, In-containment refuelling water storage tank (IRWST), and portions of the auxiliary systems. The containment building is designed to sustain all internal and external loading conditions which are reasonably expected to occur during the life of the plant. The containment building is on a common basemat which forms a monolithic structure with the auxiliary building.

The interior arrangement of the containment building is designed to meet the requirements for all anticipated conditions of operations and maintenance, including new and spent fuel handling. There are four main floor levels in the containment: the lowest floor level, called the basement, the highest floor elevation, called the operation floor, and two (2) mezzanine floors in between the basement and operating floors. The two mezzanine floors are designed primarily of steel-supported grating.

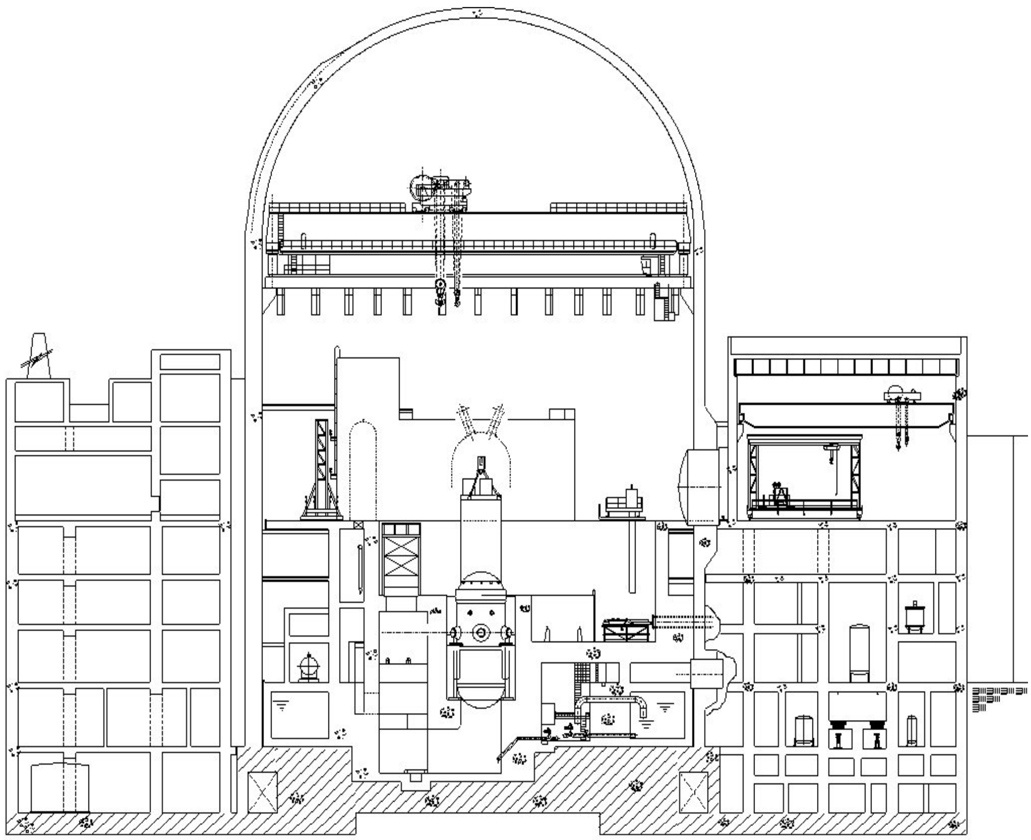


FIG. 4.9-10. Cross-sectional view of the reactor building (vertical direction)

The equipment hatch is at the operating floor level, and has an inside diameter of 7.8 m (26 feet). This hatch size is selected to accommodate the one-piece replacement of a steam generator. A polar bridge crane is supported from the containment wall. The bridge crane has the capability to install and remove the steam generators. Personnel access to the containment is through two hatches, one located at the operating floor level and the other at the plant ground elevation.

The containment is a post-tensioned concrete cylinder with an internal diameter of 45.7 m (150 ft) and a hemispherical top dome. There is no structural connection between the free standing portion of the 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.

4.9.7.4 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 configuration is simplified for constructability, and the maintainability of the systems is improved by centralizing the condensate polishing system, separating the switchgear building, and rearranging the equipment hatches. There are four main floor levels referred to as the basement, ground level, operating level, and deaerator level.

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 the failure of safety related structures. The turbine building is located such that the containment building is at the high pressure turbine side on the projection of the turbine shaft. This allows the 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. The vibration problem which occurs during transient loading was minimized by moving the fresh water tank of the steam generator blowdown system to the auxiliary building.

In the APR1400 plant, the 52 inches Last stage blade (LSB) of the LP turbine was taken into consideration for the building design. Other items reflected in the general arrangement design are as follows:

- Relocation of the TBCCW heat exchanger into the turbine building ;
- Relocation of the secondary sample room & lab to the compound building;
- Simplification of contour of the turbine building super structure.

4.9.7.5 Other buildings

Auxiliary building arrangement

The auxiliary building completely surrounds the containment building and is on the common basemat which forms a monolithic structure with the containment building. The diesel generator room is built into the auxiliary building. To assure the safety and reliability, the auxiliary building is designed to enhance physical separation for the mitigation of internal flooding and fire propagation. The auxiliary building houses pumps and heat exchangers for the safety injection system and shutdown cooling system. Also, the auxiliary feedwater tanks and main control room are located in the auxiliary building. For the convenience of operation and maintenance, there is a staging service area in the auxiliary building for installation work in front of the equipment hatch of the containment.

The emergency diesel generator (EDG) area is located in the auxiliary building at the ground level. The fuel storage tanks are located on each side of the auxiliary building. The EDGs are arranged as separate entities with dedicated auxiliaries including air supply, exhausts, and cooling systems, so that they are independent of each other in all respects. The EDG areas are arranged to provide routine maintenance facilities and maintenance access space such that work on one EDG will not affect the operability of the other EDG.

Compound building arrangement

The compound building is an integrated building of the radwaste and access control buildings. The compound building consists of an access control facility, a radwaste treatment facility, a hot machine shop, and sampling facilities & lab. The compound building is designed to be shared between two units and is classified as non-safety related.

Radiation shielding is provided wherever required. For the building arrangement design, the protection against natural phenomena and the accommodation of associated environmental conditions were reflected to retain the spillage of potentially contaminated solids or liquids within the building. It has no major structural interface with other buildings, the access control is made at the ground floor in the compound building.

Switchgear building

The switchgear building is located in the vicinity of the turbine building and all the electrical switchgears are centralized in this area for the convenience of maintenance and efficiency of space allocation.

4.9.8 Technical data

General plant data

Power plant output, gross	1 450	MWe
Reactor thermal output	4 000	MW
Total core heat output	3 983	MW
Power plant efficiency (net)	34.8	%
Plant lifetime	60	years
Seismic design, SSE	0.3	g

Nuclear steam supply system

Number of coolant loops	2	
Primary circuit volume, including pressurizer	454.7	m^3
Steam flow rate at nominal conditions per SG	1079	kg/s
Feedwater flow rate at nominal conditions	-	kg/s
Steam temperature/pressure(operating)	285/6.9	^oC/MPa
Feedwater temperature	232.2	^oC

Reactor coolant system

Primary coolant flow rate	20991	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	290.6	^oC
Coolant outlet temperature, at RPV outlet	323.9	^oC

Reactor core

Core height(active fuel)	3.75	m
Equivalent core diameter	3.65	m
Heat transfer surface in the core	6 592	m^2
Fuel inventory	228	t U
Average fuel power density	17.5	kW/kg U
Average core power density (volumetric)	0.25	kW/cm^3
Enthalpy rise, F_H	1.55	
DNB ratio	2.12	
Fuel material	Sintered UO2	
Fuel assembly total length	4 127.5	mm
Rod array	square, 16x16	
Number of fuel assemblies	241	
Number of fuel rods/assembly	236	
Number of grids per assembly	11	
Enrichment (range) of first core	1.6, 2.8/3.3, 3.3/3.8 Wt%	
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel(First core)	30 700	MWD/MTU
Cladding tube material	Zr-4	
Burnable absorber, strategy/material	UO_2-Gd_2O_3	
No. control assembly(full/part strength)	76/17	
Absorber rods per control assembly	4 or 12	
Absorber material (full/part strength)	B_4C/Inconel 625	
Drive mechanism		
Positioning rate		steps/min
Soluble neutron absorber	Boron	

Reactor pressure vessel

Total volume	162.3	m^3
Cylindrical shell inner diameter	4 630	mm
Wall thickness of cylindrical shell	230	mm
Total height	15 280	mm
Base material: cylindrical shell	SA 508, Class 2&3	
RPV head	Inconel 690	
Liner	Stainless steel	
Design pressure/temperature	17.2/343.3	MPa/^oC
Transport weight (lower part)	406.7	t
RPV head	115.4	t

Steam generators

Type	Vertical, U-tube	
Number	2	
Heat transfer surface per S/G	15 033	m^2
Number of heat exchanger tubes per S/G	12 596	
Tube dimensions (outer diameter/thickness)	19.05/1.07	mm
Moisture carryover, weight maximum	0.25	%
Maximum outer diameter	6 172	mm
Total height	22 235	mm
Transport weight	905.4	t
Shell and tube sheet material	SA 533 Grade A/B, Class 1, SA 508, Class 2 or 3	
Tube material	Inconel 690	
Tube plugging margin	10	%

Reactor coolant pump

Type	Vertical, single-stage, centrifugal pump	
Number	4	
Design pressure/temperature	17.6/343.3	MPa/°C
Rated flow rate (at operating conditions)	7 671	kg/s
Pump head	182.9	m
Power demand at coupling, cold/hot	8 948/6 711	kW
Pump casing material	SA 508, Class 2 or 3	
Pump speed	1 190	rpm

Pressurizer

Total volume	68.9	m^3
Steam volume: full power	34.9	m^3
Design pressure/temperature	17.2/371.1	MPa/ °C
Heating power of the heater rods	2 400	kW
Number of heater rods	48	
Inner diameter	2 438	mm
Total height	16 459	mm
Material	SA 533 Grade A/B, Class 1, SA 508, Class 2 or 3	
Transport weight	164.9	t

Containment

Type	Steel-lined post-tensioned prestressed concrete	
Overall form (spherical/cyl.)	Cylindrical	
Dimensions (diameter/height)	45.7/76.4	m
Free volume	90 444	m^3
Design pressure(DBEs)	480	kPa
(Severe accident situations)		kPa
Design leakage rate(24h)	0.15	vol%/day
(after 24h)	0.075	vol%/day

Engineered Safety features

Safety injection system		
High head pump, No.	4	
Low head pump, No.0		
Injection tank, No.	4	
Auxiliary feedwater system		
Pump, No.	2(motor), 2 (turbine)	
Rated flow rate, each	34.7	l/sec
Containment Safety system		
Spray pump, No.	2	
Spray heat exchanger, No.	2	
Spray pump design capacity(each)	315	l/sec
Component cooling water system		
Number of pump per division	2	
Number of heat exchanger, type	3, plate	
Design capacity, each	1 072. 5	l/sec

Power supply systems

Main transformer,	rated voltage	22.8/345	kV
	rated capacity	3x1450	MVA
Plant transformers,	rated voltage	24/14.49, 4.47	kV
	rated capacity	60/80	MVA
Unit Aux. transformer rated voltage		345/14.49, 4.47	kV
	rated capacity	61/81.4	MVA
Medium voltage busbars (6 kV or 10 kV)		13.8, 4.15	kV
Number of low voltage busbar systems		2	
Standby diesel generating units: number		2	
	rated power	7,500	kW
Number of diesel-backed busbar systems		1 per DG unit	
Voltage level of these		4.16	kV ac
Number of DC distributions		4/ 1/ 2	
Voltage level of these		125 (Class 1E)/250,125 (Non-1E)	
Number of battery-backed busbar systems		4/ 1/ 2	
Voltage level of these		125 (Class 1E)/ 250, 125 (Non-1E)	

Instrument & control

RPS reactor trip switch	2 set of 2 each in both MCR and RSP	
Automatic initiation parameter	4 channel provided,	
Channel/logic	Coincidence of 2 ch. for trip	
ESFAS, No of manual switch	2 set of 2 each ,	
Automatic initiation parameter	4 channel provided,	
Channel/logic	Coincid. of 2 ch.	

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	In-line, 6 flow, tandem regenerative reheat TC6F-52	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/ 3 LP	
Turbine speed	1800	rpm
HP inlet pressure/temperature	6.9/285	MPa/°C

Generator

Type	4-pole, 1800	rpm
Rated power	1 658.4	MVA
Voltage	24	kV
Frequency	60	Hz
Generator efficiency	99.1	%
Total generator mass(stator + rotor)	696	t
Overall length of generator	29.48	m

Condenser

Type	Once-through, sea water cooling	
Cooling water flow rate	37. 8	m^3 /s
Cooling water, inlet	32.1	^{0}C
Cooling water, outlet	< 40	^{0}C
Condenser pressure	38.1	mmHg

Condensate pumps

Number, motor driven	50% x 3	
Flow rate	37 854	kg/s
Temperature	35/51.1	°C
Pump speed	1 180	rpm

Feedwater pumps

Number(turbine driven)	50% x 3	
Speed	Variable	rpm
Rated flow rate, normal	908	l/sec
Booster pump No.(Motor driven)	50% x 3	
Startup feedwater pump		
Number, type	1, motor	
Rated flow	126	l/sec

Condensate and feedwater heaters

Number of heating stages	7	
Redundancy	3 strings, 3 per string	

4.9.9 Measures to enhance economy and maintainability

APR1400 has economical competitiveness with substantial cost savings over the currently operating Korean nuclear power plants. Several factors affecting the economics have been thoroughly reviewed from the beginning of project. It incorporated a number of design improvements to meet the Utility Requirement Documents (URDs) for the economic goals (See Table 4.9-1). The goals were to increase the power level, to reduce the construction duration, to improve the availability factors, and to simplify the system and structural design based on the experiences of construction, operation and maintenance of Korean Standard Nuclear Power Plants (KSNPs). Especially, at the end of the basic design phase, a thorough review of the design was carried out jointly with the design development and construction teams to reinforce the economic competence and constructability of APR1400. As a result of this effort, therefore, the design optimization of the systems and general arrangement was implemented.

In the optimization processes of APR1400, the double containment and Passive Secondary Condensing System (PSCS) were modified or eliminated after the cost-benefit analyses. Even if the double containment was effective for the safety enhancement, the constructability and maintainability of the double containment were found to be very costly. Since APR1400 is equipped with the external reactor vessel cooling system for severe accident mitigation, the change of the containment type from the double to single containment was found to make insignificant impact on achieving the safety goals. Also, Passive Secondary Condensing System (PSCS) - a passively operated backup system in case of Auxiliary Feedwater System failure - which was adopted in the early design of APR1400, was found to be less cost effective. and enhancing the reliability of the auxiliary feedwater pumps would be more cost effective. Therefore, the PSCS was removed during the optimization process.

With the objective of 5 ~ 10% of volume and bulk material reduction in the general arrangement, access control building, rad-waste building, primary and secondary sampling room and hot machine shop were incorporated into the compound building which was also shared between the two units. This twin unit concept with the shared facilities between the two units helps the economic competence of the APR1400. The integrated design of the emergency diesel generator and fuel buildings into the auxiliary building were also implemented.

The design features incorporated for the operating convenience and maintenance flexibility would help enhance the economic competitiveness of APR1400. For example, the improvement of operator's recognition for plant situation according to the use of workstation and large display panel as well as using computerized system and operation information display on graphic screen will contribute to the operational flexibility. Also, the containment layout enabling the one piece removal of a steam generator, Integrated Head Assembly (IHA) will shorten the outage duration, resulting in the higher availability. The addition of a platform for S/G In-Service Inspection (ISI) and RCP maintenance will also help the operational and maintenance flexibility. Also, Inconel-690 and lower hot leg temperature were adopted to prolong the integrity of the S/G tubes and expected to result in the lifetime economics eventually.

The construction duration of nuclear power plant affects the economics of nuclear power because the capital cost is generally significant. Accordingly, it is important to develop a design which can reduce the construction period. In the APR1400 design process, a 48-month and a 54-month construction schedules from the first concrete placement to the commercial operation were developed and compared to the actual accomplishments of KSNPs which were ranging from 58 to 64 months. Various construction methods were studied and suggested such as over-the-top method and deck plate method. The former was recommended for the installation of NSSS components and the latter was for the installation of mechanical and electrical equipment to reduce the auxiliary building construction. Even if the 48 month schedule for the N-th plant was a very ambitious goal, it was considered a possible way with the proper application of construction techniques.

The economics assessment indicated that the availability of the plant was the most sensitive parameter for ensuring the validity of the cost estimate of electricity production. The availability improvement can be achieved by the reduction of both forced outage and planned outage time (normally for refueling purpose). Analytical assessment for the system and component availability was performed to reduce the number of the forced outages by enhancing the system reliability. To reduce the planned outage duration, the APR1400 standard outage schedule was generated by reviewing the experiences from KSNP. The overall availability factor including normal, extended and forced outage was estimated to be more than 90% which met the project goal.

In summary, the APR1400's electric power was increased by 40% compared with that of KSNPs while the material quantity of APR1400 was estimated to be increased by 20 %. The evaluation showed that the APR1400 would be economically superior to coal-fired power plants, i. e., about a 20% cost competitiveness against a 500MWe class coal-fired power plant is expected.

TABLE 4.9-II. MEASURES IMPLEMENTED IN APR1400 FOR ECONOMIC IMPROVEMENT

Items	Description
Simplification /Optimization	- Elimination of Passive Secondary Condensing System (PSCS) - Adoption of single containment (outer containment removed) - A shared compound building for two units - Removal of non-generative heat exchanger from the SGBD system - Elimination of TBOCW pump and function incorporated into CW pump - Elimination of steam jet air ejector and function incorporated into condenser - Removal of common header in safety injection system and low pressure safety injection pump - Integration of fuel handling building and emergency diesel generator building
Operational convenience and Maintenance flexibility	- Removal of recirculation operation mode in SIS - Arrangement of primary and secondary sample room in the compound building - Adoption of Integrated Head Assembly(IHA) - Design of a crane for RCP seal repairing - Design improvement of SG room platform and access port
Constructibility	- Application of new construction method • Over-the-Top construction method • Modular construction method • Deck plate construction method - Application of new technology • Multi-signal cable to reduce the quantity of cable/cable tray • Constructibility through 3-D modeling • Auto-welding of reactor coolant pipe
Performance / Economic Improvement	- Long term refueling cycle of 18 ~ 24 months • Increase plant capacity factor • Reduce the amount of spent fuel treatment • Reduce radiation dose • Reduce O&M cost - Security of legal certification of standard design • Cost reduction in design, equipment manufacturing, installation by repeated construction

4.9.10 Project status and planned schedule

The development of APR1400 standard design was launched at the end of 1992, organized in three phases related to the development status. The third phase ended at December 2001 and the design certification was issued May 2002 by Korean regulatory agency.

Phase I was finished for two years and the major activity was to develop the top tier design requirements and concepts for the new design. Phase II was a four-year program running from 1995 to 1998. The major activities of this phase were to develop a basic design for a licensing review, to ensure the safety of the APR1400 and thus, its licensibility.

The main activity of Phase III was the licensing review and licensing support works for the standard design certification. Also, continuous design development for the long-lead items such as the advanced main control room design with digitized I&C and major NSSS components was performed in parallel with the design optimization described in 4.9.9. During the APR1400 licensing review, there were more than 2,000 Request for additional informations (RAIs) from regulatory agency. Also, major advanced features were verified through experiments and mock-up tests. They are the direct vessel injection experiments, the sparger test for condensation in the IRWST, the performance test of the fluidic device, the fatigue test for the CEDM design, and the human factors verification for the main control room design using the dynamic mockup.

According to the long-term power development program in the Republic of Korea, two units of APR1400 are scheduled for operation in September 2010 and September 2011 respectively. The construction plan of these units was set up and the site for the first APR1400 is prepared in Shin-Kori, which is located at the southeast shore of the Republic of Korea. In the construction plan of the first of the two units, the first concrete will be poured in June 2005, and the commissioning will be made in September 2010. When APR1400 is in commercial operation in 2010, it is expected to be the first evolutionary type advanced PWR plant generating electricity.

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4.10 AP-1000 (WESTINGHOUSE, USA)

4.10.1 Introduction

The Westinghouse Advanced Passive PWR AP1000 is a 1117 MWe PWR based closely on the AP600 design. The AP1000 maintains the AP600 design configuration, use of proven components and licensing basis by limiting the changes to the AP600 design to as few as possible. The AP1000 design includes advanced passive safety systems and extensive plant simplifications to enhance the safety, construction, operation, and maintenance of the plant. The plant design utilizes proven technology, which builds on approximately 40 years of operating PWR experience. PWRs represent 74 percent of all Light Water Reactors around the world, and the majority of these are based on Westinghouse PWR technology.

The AP1000 is designed to achieve a high safety and performance record. It is conservatively based on proven PWR technology, but with an emphasis on safety features that rely on natural forces. Safety systems use natural driving forces such as pressurized gas, gravity flow, natural circulation flow, and convection. Safety systems do not use active components (such as pumps, fans or diesel generators) and are designed to function without safety-grade support systems (such as AC power, component cooling water, service water, HVAC). The number and complexity of operator actions required to control the safety systems are minimized; the approach is to eliminate operator action rather than automate it.

The AP1000 is designed to meet U.S. NRC deterministic safety criteria and probabilistic risk criteria with large margins. Safety analysis has been completed and documented in the Design Control Document (DCD) and Probabilistic Risk Analysis (PRA). The extensive AP600 testing program, which is applicable to the AP1000, verifies that the plant's innovative features will perform as designed and analyzed. PRA results show a very low core damage frequency, which meets the goals established for advanced reactor designs and a low frequency of release due to improved containment isolation and cooling.

An important aspect of the AP1000 design philosophy focuses on plant operability and maintainability. The AP1000 design includes features such as simplified system design to improve operability while reducing the number of components and associated maintenance requirements. In particular, simplified safety systems reduce surveillance requirements by enabling significantly simplified technical specifications.

Selection of proven components has been emphasized to ensure a high degree of reliability with a low maintenance requirement. Component standardization reduces spare parts, minimizes maintenance, training requirements, and allows shorter maintenance durations. Built-in testing capability is provided for critical components.

Plant layout ensures adequate access for inspection and maintenance. Laydown space provides for staging of equipment and personnel, equipment removal paths, and space to accommodate remotely operated service equipment and mobile units. Access platforms and lifting devices are provided at key locations, as are service provisions such as electrical power, demineralized water, breathing and service air, ventilation and lighting.

The AP1000 design also incorporates radiation exposure reduction principles to keep worker dose as low as reasonably achievable (ALARA). Exposure length, distance, shielding and source reduction are fundamental criteria that are incorporated into the design.
Various features incorporated in the design to minimize construction time and total cost by eliminating components and reducing bulk quantities and building volumes include:

- Flat, common Nuclear Island basemat design minimizes construction cost and schedule;
- Integrated protection system, advanced control room, distributed logic cabinets, multiplexing, and fiber optics, significantly reduce the quantity of cables, cable trays, and conduits;
- Stacked arrangement of the Class 1E battery, dc switchgear, integrated protection system, and the main control rooms eliminate the need for the upper and lower cable spreading rooms that are required in current generation PWR plants;
- Application of the passive safety systems replaces and/or eliminates many of the conventional mechanical safety systems typically located in Seismic Category I buildings in current generation PWR plants.

The AP1000 is designed with environmental consideration as a priority. The safety of the public, the power plant workers, and the impact to the environment have been addressed as follows:

- Operational releases have been minimized by design features;
- Aggressive goals for worker radiation exposure have been set and satisfied;
- Total radwaste volumes have been minimized;
- Other hazardous waste (non-radioactive) have been minimized.

The AP1000 has a well-defined design basis that has been confirmed through thorough engineering analyses and testing. Some of the high-level design characteristics of the plant are:

- Net electrical power of at least 1117 MWe; and a thermal power of 3415 MWt;
- Rated performance is achieved with up to 10% of the steam generator tubes plugged and with a maximum hot leg temperature of 610°F (321°C);
- Core design is robust with at least a 15% operating margin on core power parameters;
- Short lead time (five years from owner's commitment to commercial operation) and construction schedule (3 years);
- No plant prototype is needed since proven power generating system components are used;
- Major safety systems are passive; they require no operator action for 72 hours after an accident, and maintain core and containment cooling for a protracted time without ac power;
- Predicted core damage frequency of 2.4E-07/yr is well below the 1E-05/yr requirement, and frequency of significant release of 1.95E-08/yr is well below the 1E-06/yr requirement;
- Standard design is applicable to anticipated sites in the U.S. and in other countries;
- Occupational radiation exposure expected to be below 0.7 man-Sv/yr (70 man-rem/yr);
- Core is designed for a 18-month fuel cycle;
- Refueling outages can be conducted in 17 days or less;
- Plant design life of 60 years without replacement of the reactor vessel;
- Overall plant availability greater than 93%, including forced and planned outages; the goal for unplanned reactor trips is less than one per year;
- Leak-before-break on primary lines > 6-inches and on main steam lines;
- Seismic based on 0.3g ground acceleration;
- Security enhanced with all safe shutdown equipment located in safety reinforced concrete Nuclear Island buildings;
- Meets URD and EUR requirements;
- Invessel retention of core debris following core melt which significantly reduces the uncertainty in the assessment of containment failure and radioactive release to the environment due to ex-vessel severe accident phenomena;
- No reactor pressure vessel penetrations below the top of the core. This eliminates the possibility of a loss of coolant accident by leakage from the reactor vessel, which could lead to core uncovery.

4.10.2 Description of the nuclear systems

4.10.2.1 Primary circuit and its main characterisitcs

The primary circuit of the AP1000 reactor retains most of the general design features of current designs, with added evolutionary features to enhance the safety and maintainability of the system. The system consists of two heat transfer circuits (Figure 4.10-1) each with a single hot leg and two cold legs, a steam generator, and two reactor coolant pumps installed directly onto the steam generator; eliminating the primary piping between pumps and steam generator. A simplified support structure for the primary systems reduces in-service inspections and improves accessibility for maintenance.

The reactor coolant system pressure boundary provides a barrier against release of radioactivity and is designed to provide a high degree of integrity throughout operation of the plant.

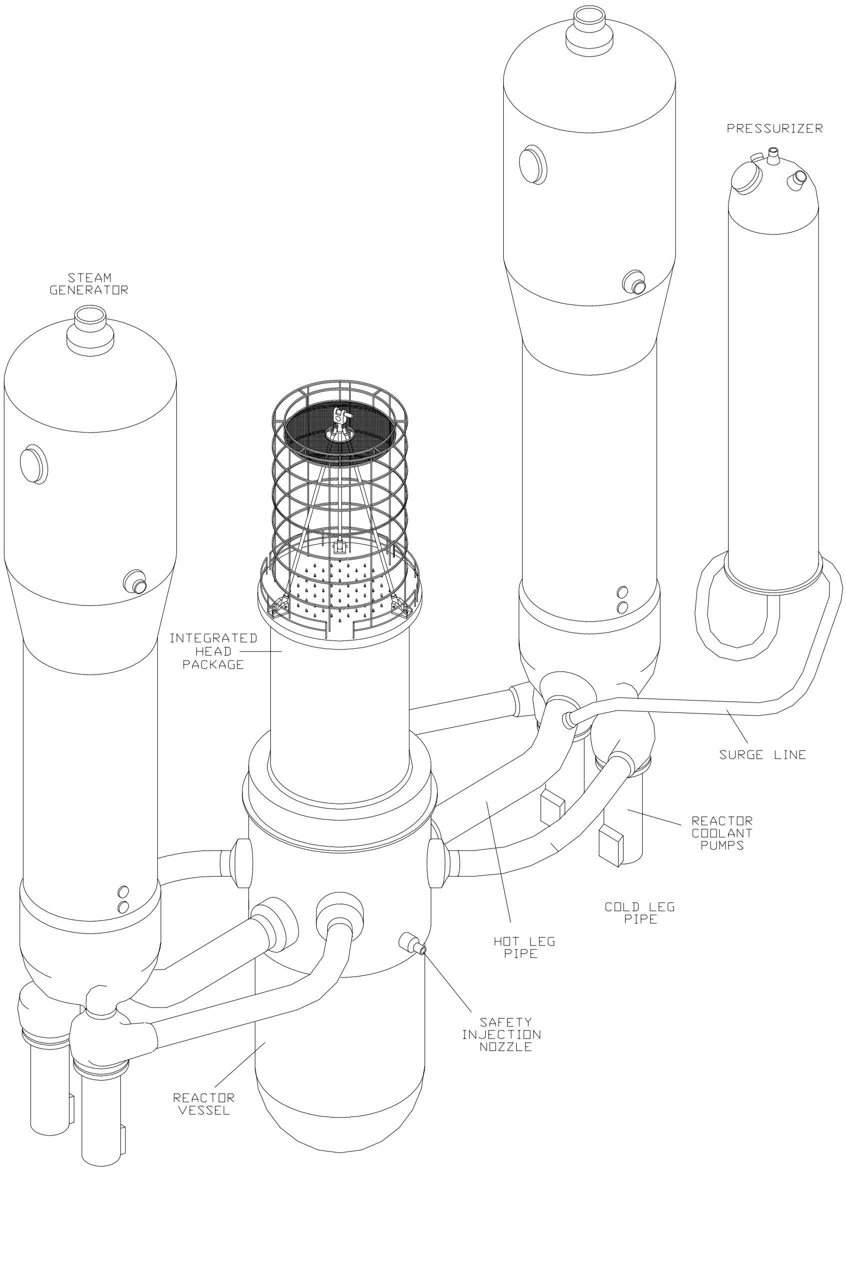


FIG. 4.10-1 Isometric view of AP1000 NSSS

4.10.2.2 *Reactor core and fuel design*

The core, reactor vessel, and reactor internals of the AP1000 are similar to those of conventional Westinghouse PWR designs. Several important enhancements, all based on existing technology, have been used to improve the performance characteristics of the design. The AP1000 incorporates a low boron core design to increase safety margins for accident scenarios such as Anticipated Transients Without Scram. Fuel performance improvements include ZIRLO™ grids, removable top nozzles, and longer burnup features. The AP1000 core incorporates the Westinghouse ROBUST fuel assembly design compared to the Vantage 5-H design of the AP600. The reactor core is comprised of 157, 14 foot (426.7 cm), 17×17 fuel assemblies. The AP1000 core design provides a robust design with at least 15 percent in departure from nucleate boiling (DNB) margin.

The materials that serve to attenuate neutrons originating in the core and gamma rays from both the core and structural components consist of the core shroud, core barrel and associated water annuli. These are within the region between the core and the pressure vessel.

The core consists of three radial regions that have different enrichments; the enrichment of the fuel ranges from 2.35 to 4.8%. The temperature coefficient of reactivity of the core is highly negative. The core is designed for a fuel cycle of 18 months with a 93% capacity factor, region average discharge burnups as high as 60000 MWd/t.

The AP1000 uses reduced-worth control rods (termed "gray" rods) to achieve daily load follow without requiring changes in the soluble boron concentration. The use of gray rods, in conjunction with an automated load follow control strategy, eliminates the need for processing thousands of gallons of water per day to change the soluble boron concentration. As a result, systems are simplified through the elimination of boron processing equipment (such as evaporator, pumps, valves, and piping). With the exception of the neutron absorber materials used, the design of the gray rod assembly is identical to that of a normal control rod assembly.

4.10.2.3 *Fuel handling and transfer systems*

Refueling is performed in the same way as for current plants. After removing the vessel head, fuel handling takes place from above, using the refueling machine to configure the core for the next cycle.

New fuel storage - New fuel is stored in a high-density rack which includes integral neutron absorbing material to maintain the required degree of sub-criticality. The rack is designed to store fuel of the maximum design basis enrichment. The new fuel rack includes storage locations for 72 fuel assemblies. Minimum separation between adjacent fuel assemblies is sufficient to maintain a sub-critical array even in the event the building is flooded with unborated water, fire extinguishing aerosols, or during any design basis event.

Spent fuel storage - Spent fuel is stored in high density racks which include integral neutron absorbing material to maintain the required degree of sub-criticality. The racks are designed to store fuel of the maximum design basis enrichment. The spent fuel storage racks include storage locations for 619 fuel assemblies. The modified 10×7 rack module additionally contains integral storage locations for five defective fuel assemblies. The design of the rack is such that a fuel assembly can not be inserted into a location other than a location designed to receive an assembly.

4.10.2.4 Primary components

Reactor pressure vessel – The reactor vessel (Figure 4.10-2) is the high-pressure containment boundary used to support and enclose the reactor core. The vessel is cylindrical, with a hemispherical bottom head and removable flanged hemispherical upper head.

The reactor vessel is approximately 39.5 feet (12.0 m) long and has an inner diameter at the core region of 157 inches (3.988 m). Surfaces, which can become wetted during operation and refueling, are clad with stainless steel welded overlay. The AP1000 reactor vessel is designed to withstand the design environment of 2500 psia (17.1 MPa) and 650°F (343°C) for 60 years. As a safety enhancement, there are no reactor vessel penetrations below the top of the core. This eliminates the possibility of a loss of coolant accident (LOCA) by leakage from the reactor vessel, which could lead to core uncovery. The core is positioned as low as possible in the vessel to limit reflood time in accident situations.

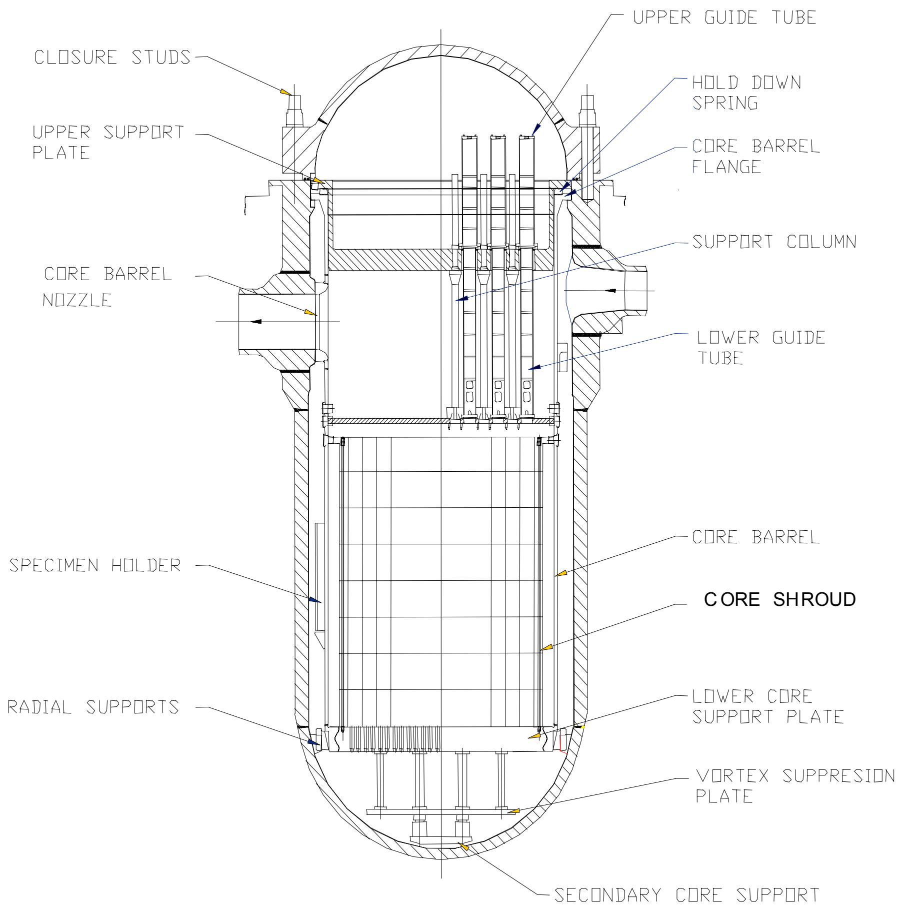


FIG. 4.10-2 AP1000 Reactor pressure vessel

Reactor internals - The reactor internals, the core support structures, the core shroud, the downcomer and flow guiding structure arrangement, and the above-core equipment and structures, are very similar to those in current plants.

The reactor internals consist of two major assemblies - the lower internals and the upper internals. The reactor internals provide the protection, alignment and support for the core, control rods, and gray rods to provide safe and reliable reactor operation.

Steam generators - Two model Delta-125 steam generators (Figure 4.10-3) are used in the AP1000 plant. The high reliability of the steam generator design is based on design enhancements and a proven design. The steam generator design is based on the following proven designs: Delta-75 replacement steam generators for V.C. Summer and other plants; Delta-94 replacement steam generator for South Texas plant; Replacement steam generators (1500 MWt per SG) for Arkansas (ANO); San Onofre and Waterford steam generator designs with capacities similar to the AP1000 steam generators. The steam generators operate on all volatile treatment secondary side water chemistry. Steam generator design enhancements include full-depth hydraulic expansion of the tubes in the tubesheets, nickel-chromium-iron Alloy 690 thermally treated tubes on a triangular pitch, broached tube support plates, improved anti-vibration bars, upgraded primary and secondary moisture separators, enhanced maintenance features, and a primary-side channel head design that allows for easy access and maintenance by robotic tooling. All tubes in the steam generator are accessible for sleeving, if necessary.

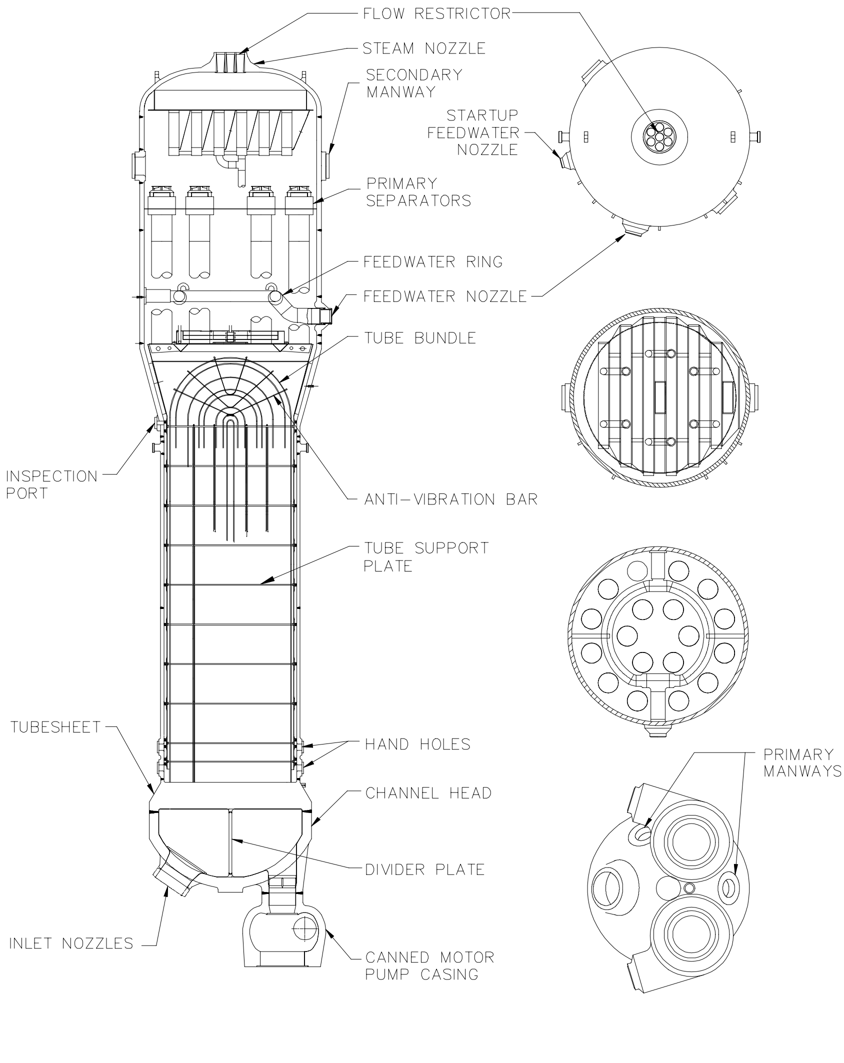


FIG. 4.10-3 AP1000 Steam generator

Pressurizer - The AP1000 pressurizer is of conventional design, based on proven technology. The pressurizer volume is 2100 ft^3 (59.5 m^3). The large pressurizer avoids challenges to the plant and operator during transients, which increases transient operation margins resulting in a more reliable plant with fewer reactor trips. It also eliminates the need for fast-acting power-operated relief valves, a possible source of RCS leakage and maintenance.

Reactor coolant pumps - These pumps are high-inertia, highly-reliable, low-maintenance, hermetically sealed canned-motor pumps that circulate the reactor coolant through the reactor core, loop piping, and steam generators. The AP1000 pump is based on the AP600 canned-motor pump design with provisions to provide more flow and a longer flow coast down. The motor size is minimized through the use of a variable speed controller to reduce motor power requirements during cold coolant conditions. Two pumps are mounted directly in the channel head of each steam generator. This configuration eliminates the cross-over leg of coolant loop piping; reduces the loop pressure drop; simplifies the foundation and support system for the steam generator, pumps, and piping; and reduces the potential for uncovering of the core by eliminating the need to clear the loop seal during a small LOCA. The reactor coolant pumps have no seals, eliminating the potential for seal failure LOCA, which significantly enhances safety and reduces pump maintenance. The pumps use a flywheel to increase the pump rotating inertia. The increased inertia provides a slower rate-of-flow coastdown to improve core thermal margins following the loss of electric power. Testing has validated the manufacturability and operability of the pump flywheel assembly.

Main coolant lines - Reactor coolant system (RCS) piping is configured with two identical main coolant loops, each employing a single 31-inch (790 mm) inside 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 22-inch (560 mm) inside diameter cold leg pipes in each loop (one per pump) transport reactor coolant back to the reactor vessel to complete the circuit.

The RCS loop layout contains several important features that provide for a significantly simplified and safer design. The reactor coolant pumps mount directly on the channel head of each steam generator, which allows the pumps and steam generator to use the same structural support, greatly simplifying the support system and providing more space for maintenance. The combined steam generator/pump vertical support is a single pinned column extending from the floor to the bottom of the channel head. The steam generator channel head is a one-piece forging with manufacturing and inspection advantages over multipiece, welded components. The integration of the pump suction into the bottom of the steam generator channel head eliminates the crossover leg of coolant loop piping, thus avoiding the potential for core uncovery due to loop seal venting during a small LOCA.

The simplified, compact arrangement of the RCS also provides other benefits. The two cold leg lines of the two main coolant loops are identical (except for instrumentation and small line connections) and include bends to provide a low-resistance flow path and flexibility to accommodate the expansion difference between the hot and cold leg pipes. The piping is forged and then bent, which reduces costs and in-service inspection requirements. The loop configuration and material selection yield sufficiently low pipe stresses so that the primary loop and large auxiliary lines meet leak-before-break requirements. Thus, pipe rupture restraints are not required, greatly simplifying the design and providing enhanced access for maintenance. The simplified RCS loop configuration also allows for a significant reduction in the number of snubbers, whip restraints, and supports. Field service experience and utility feedback have indicated the high desirability of these features.

4.10.2.5 Reactor auxiliary systems

Chemical and volume control system – The chemical and volume control system (CVS) consists of regenerative and letdown heat exchangers, demineralizers and filters, makeup pumps, tanks, and associated valves, piping, and instrumentation, and is designed to perform the following major functions:

- **Purification** - maintain reactor coolant purity and activity level within acceptable limits;
- **Reactor coolant system inventory control and makeup** - maintain the required coolant inventory in the reactor coolant system; maintain the programmed pressurizer water level during normal plant operations;
- **Chemical shim and chemical control** - maintain reactor coolant chemistry during plant startups, normal dilution to compensate for fuel depletion and shutdown boration and provide the means for controlling the reactor coolant system pH by maintaining the proper level of lithium hydroxide;
- **Oxygen control** - provide the means for maintaining the proper level of dissolved hydrogen in the reactor coolant during power operation and for achieving the proper oxygen level prior to startup after each shutdown;
- **Filling and pressure testing of the reactor coolant system** - provide the means for filling and pressure testing of the reactor coolant system. The chemical and volume control system does not perform hydrostatic testing of the reactor coolant system, but provides connections for a temporary hydrostatic test pump;
- **Borated makeup to auxiliary equipment** - provide makeup water to the primary side systems, which require borated reactor grade water;
- **Pressurizer Auxiliary Spray** - provide pressurizer auxiliary spray water for depressurization.

Normal residual heat removal system - This system consists of two mechanical trains of equipment, each comprising one pump and one heat exchanger. The two trains of equipment share a common suction line from the reactor coolant system and a common discharge header. The system includes the piping, valves and instrumentation necessary for system operation. The major functions are:

- **Shutdown heat removal -** remove residual and sensible heat from the core and the reactor coolant system during plant cooldown and shutdown operations. The system provides reactor coolant system cooldown from 350 to 120°F (177 to 48.9°C) within 96 hours after shutdown. The system maintains the reactor coolant temperature at or below 120°F during plant shutdown.
- **Shutdown purification -** provide reactor coolant system and refuelling cavity purification flow to the chemical and volume control system during refuelling operations.
- **In-containment refueling water storage tank cooling** – provide cooling to the IRWST to limit the IRWST water temperature to less than 212°F (100°C) during extended operation of the passive residual heat removal system and to not greater than 120°F during normal operation.
- **Low pressure reactor coolant system makeup and cooling** - provide low pressure makeup from the cask loading pit and then the IRWST to the reactor coolant system and provide additional margin for core cooling.
- **Low temperature overpressure protection** - provide low temperature overpressure protection for the reactor coolant system during refuelling, startup, and shutdown operations.
- **Long-term, post-accident containment inventory makeup flowpath** - provide a flow path for long term post-accident makeup to the reactor containment inventory, under design assumptions of containment leakage.
- **Post-accident recovery** - Remove heat from the core and the reactor coolant system following successful mitigation of an accident by the passive core cooling system
- **Spent fuel pool cooling** - Provide backup for cooling the spent fuel pool.

Spent Fuel Pool Cooling System - This system is designed to remove decay heat which is generated by stored fuel assemblies from the water in the spent fuel pool. This is done by pumping the high temperature water from within the fuel pool through a heat exchanger, and then returning the water to the pool. A secondary function of this system is clarification and purification of the water in the spent fuel pool, the transfer canal, and the refueling water. The major functions of the system are:

- **Spent fuel pool cooling** - Remove heat from the water in the spent fuel pool during operation to maintain the pool water temperature within acceptable limits.
- **Spent fuel pool purification** - Provide purification and clarification of the spent fuel pool water during operation.
- **Refueling cavity purification** - Provide purification of the refueling cavity during refueling operations.
- **Water transfers** - Transfer water between the in-containment refueling water storage tank (IRWST) and the refueling cavity during refueling operations.
- **In-containment refueling water storage tank purification** - Provide purification and cooling of the IRWST during normal operation.

4.10.2.6 Operating characteristics

The plant control scheme is based on the "reactor follows plant loads". A grid fluctuation can be compensated for through turbine control valves in case of a frequency drop. A decrease in pressure at the turbine would require an increase in reactor power.

The AP1000 is designed to withstand the following operational occurrences without the generation of a reactor trip or actuation of the safety related passive engineered safety systems. The logic and setpoints for the Nuclear Steam Supply System control systems are developed in order to meet the following operational transients without reaching any of the protection system setpoints.

- ·± 5%/minute ramp load change within 15% and 100% power
- ± 10% step load change within 15% and 100% power
- 100% generator load rejection
- 100-50-100% power level daily load follow over 90% of the fuel cycle life
- Grid frequency changes equivalent to 10% peak-to-peak power changes at 2%/minute rate
- 20% power step increase or decrease within 10 minutes
- Loss of a single feedwater pump

4.10.3 Description of turbine generator plant system

4.10.3.1 Turbine generator plant

The AP1000 turbine is a power conversion system designed to change the thermal energy of the steam flowing through the turbine into rotational mechanical work, which rotates a generator to provide electrical power. The AP1000 turbine consists of a double-flow, high-pressure cylinder and three double-flow, low-pressure cylinders that exhaust to individual condensers. It is a six flow tandem-compound, 1800 rpm machine (1500 rpm for 50 HZ applications). The turbine generator is intended for base load operation but also has load follow capability. The mechanical design of the turbine root and rotor steeple attachments uses optimized contour to significantly reduce operational stresses. Steam flow to the high-pressure turbine is controlled by two floor-mounted steam chests. Each contains two throttle/stop valve assemblies, and two load-governing valves.

The condenser and circulating water systems have been optimized. The condenser is a three-shell, multipressure unit with one double-flow, low-pressure turbine exhausting into the top of each shell.

The turbine-generator and associated piping, valves, and controls are located completely within the turbine building. There are no safety-related systems or components located within the turbine building. The probability of destructive overspeed condition and missile generation, assuming the recommended inspection frequency, is less than 1×10^{-5} per year. Turbine orientation minimizes potential interaction between turbine missiles and safety-related structures and components. The turbine-generator components and instrumentation associated with turbine-generator overspeed protection are accessible under operating conditions.

The single direct-driven generator is gas and water-cooled and rated at 1250 MVA at 24 kV, and a power factor of 0.9. Other related system components include a complete turbine-generator bearing lubrication oil system, a digital electrohydraulic (DEH) control system with supervisory instrumentation, a turbine steam sealing system, overspeed protective devices, turning gear, a generator hydrogen and seal oil system, a generator CO_2 system, an exciter cooler, a rectifier section, an exciter, and a voltage regulator.

4.10.3.2 Condensate and feedwater systems

This system supplies the steam generators with heated feedwater in a closed steam cycle using regenerative feedwater heating. The system is composed of the condensate system, the main and startup feedwater system, and portions of the steam generator system. The condensate system collects condensed steam from the condenser and pumps condensate forward to the deaerator. The feedwater system takes suction from the deaerator and pumps feedwater forward to the steam generator system utilizing high-pressure main feedwater pumps. The steam generator system contains the safety-related piping and valves that deliver feedwater to the steam generators. The condensate and feedwater systems are located within the turbine building and the steam generator system is located in the auxiliary building and containment. The main feedwater system includes three 33-1/3% single speed motor driven feedwater pumps which operate in parallel and take suction from the associated feedwater booster pumps. The discharge from the main feedwater pumps is supplied to the high-pressure feedwater heater and then to the steam generator system.

The feedwater cycle consists of six stages of feedwater heating with three parallel string, stage 1 and 2 low-pressure feedwater heaters located in the condenser neck with the next two parallel string, stage 3 and 4 low-pressure heaters, deaerator, and the two stage 6 high-pressure heaters located within the turbine building. The condenser hotwell and deaerator storage capacity allows margin in the design. This margin, coupled with three 50 percent condensate pumps, provides greater flexibility and the ability for an operator to control feedwater and condensate transients.

4.10.3.3 Auxiliary systems

*Radioactive waste management*The radioactive waste management systems include systems, which deal with liquid, gaseous and solid waste, which may contain radioactive material. The systems for liquid wastes include:

- Steam generator blowdown processing system
- Radioactive waste drain system
- Liquid radwaste system

The waste processing systems are closely integrated with the chemical and volume control system (CVS). The steam generator blowdown processing system controls and maintains the steam generator secondary cycle water chemistry. The blowdown is normally recycled to the condenser via an electronic ion exchange system, but in the case of high radiation the blowdown would be directed to the liquid radwaste system (WLS). This allows a large simplification in the blowdown system without an increase in the amount of WLS equipment.

The WLS uses ion exchangers to process and discharge all wastes from the reactor coolant system. To enhance ion exchange performance, the WLS is divided into two reprocessing trains to separate borated reactor coolant from mixed liquid waste. Based on conservative fuel defect levels and ion exchange performance consistent with the Utility Requirements Document, no evaporators are required.

A simple, vacuum-type degasifier is used to remove radioactive gases in the liquid discharge from the RCS to the WLS. The degasifier eliminates the need for cover gases or a diaphragm in the waste holdup tanks.

The gaseous radwaste system is a once-through, ambient-temperature, charcoal delay system. The system consists of a drain pot, a gas cooler, a moisture separator, an activated charcoal-filled guard bed, and two activated charcoal-filled delay beds. Also included in the system are an oxygen analyzer subsystem and a gas sampling subsystem. The radioactive fission gases entering the system are carried by hydrogen and nitrogen gas. The primary influent source is the liquid radwaste system degasifier. The degasifier extracts both hydrogen and fission gases from the chemical and volume control system letdown flow.

The solid waste management system is designed to collect and accumulate spent ion exchange resins and deep bed filtration media, spent filter cartridges, dry active wastes, and mixed wastes generated as a result of normal plant operation, including anticipated operational occurrences. The system is located in the auxiliary and radwaste buildings. Processing and packaging of wastes are by mobile systems in the auxiliary building loading bay and the mobile systems facility which is a part of the radwaste building. The packaged waste is stored in the annex, auxiliary and radwaste buildings until it is shipped offsite to a licensed disposal facility.

4.10.4 Instrumentation and control systems

The I&C system design for AP1000 integrates individual systems using similar technology. The heart of the system is the portion used for plant protection and for operation of the plant.

The integrated AP1000 I&C system provides the following benefits:

- Control wiring is reduced by 80 percent
- Cable spreading rooms are eliminated
- Maintenance is simplified
- Plant design changes have little impact on I&C design
- Accurate, drift-free calibration is maintained
- Operating margins are improved.

The AP1000 man-machine interfaces have been simplified compared to existing plants. The probability of operator error is reduced and operations, testing, and maintenance are simplified. An automatic signal selector in the control system selects from a redundant sensor for control inputs in lieu of requiring manual selection by the control board operator. Accident monitoring and safety parameters are displayed on safety qualified displays with a coordinated set of graphics generated by the qualified data processor. The major benefits of the improved man-machine interfaces are:

- Reduced quantity of manual actions is required
- Reduced quantity of data is presented to operator
- Number of alarms is reduced
- Improved quality of data is presented to operator
- Data is interpreted for the operator by system computer
- Maintenance is simplified.

4.10.4.1 Design concept, including control room

The AP1000 instrumentation and control architecture is arranged in a hierarchical manner to provide a simplified structured design that is horizontally and vertically integrated.

Above the monitor bus are the systems that facilitate the interaction between the plant operators and the I&C. These are the operations and control centers system (OCS) and the data display and monitoring system (DDS). Below the monitor bus are the systems and functions that perform the protective, control, and data monitoring functions. These are the protection and safety monitoring system (PMS) (Section 4.10.4.2) the plant control system (PLS), the special monitoring system (SMS), and the in-core instrumentation system (IIS).

The PLS has the function of establishing and maintaining the plant operating conditions within prescribed limits. The control system improves plant safety by minimizing the number of situations for which protective response is initiated and it relieves the operator from routine tasks.

The purpose of the diverse actuation system (DAS) is to provide alternative means of initiating the reactor trip and emergency safety features. The hardware and software used to implement the DAS are different from the hardware and software used to implement the protection and safety monitoring system. The DAS is included to meet the anticipated transient without (reactor) trip (ATWT) rule and to reduce the probability of a severe accident resulting from the unlikely coincidence of a transient and common mode failure of the protection and safety monitoring. The protection and safety monitoring system is designed to prevent common mode failures; however, in the low-probability case of a common mode failure, the DAS provides diverse protection.

Main control room - The operations and control centers system includes the complete operational scope of the main control room, the remote shutdown workstation, the waste processing control room, and partial scope for the technical support center. With the exception of the control console structures, the equipment in the control room is part of the other systems (for example, protection and safety monitoring system, plant control system, data and display processing system). The conceptual arrangement of the main control room is shown in Figure 4.10-4.

The boundaries of the operations and control center system for the main control room and the remote shutdown workstation are the signal interfaces with the plant components. These interfaces are via the plant protection and safety monitoring system processor and logic circuits, which interface with the reactor trip and engineered safety features plant components; the plant control system processor and logic circuits, which interface with the non-safety-related plant components; and the plant monitor bus, which provides plant parameters, plant component status, and alarms.

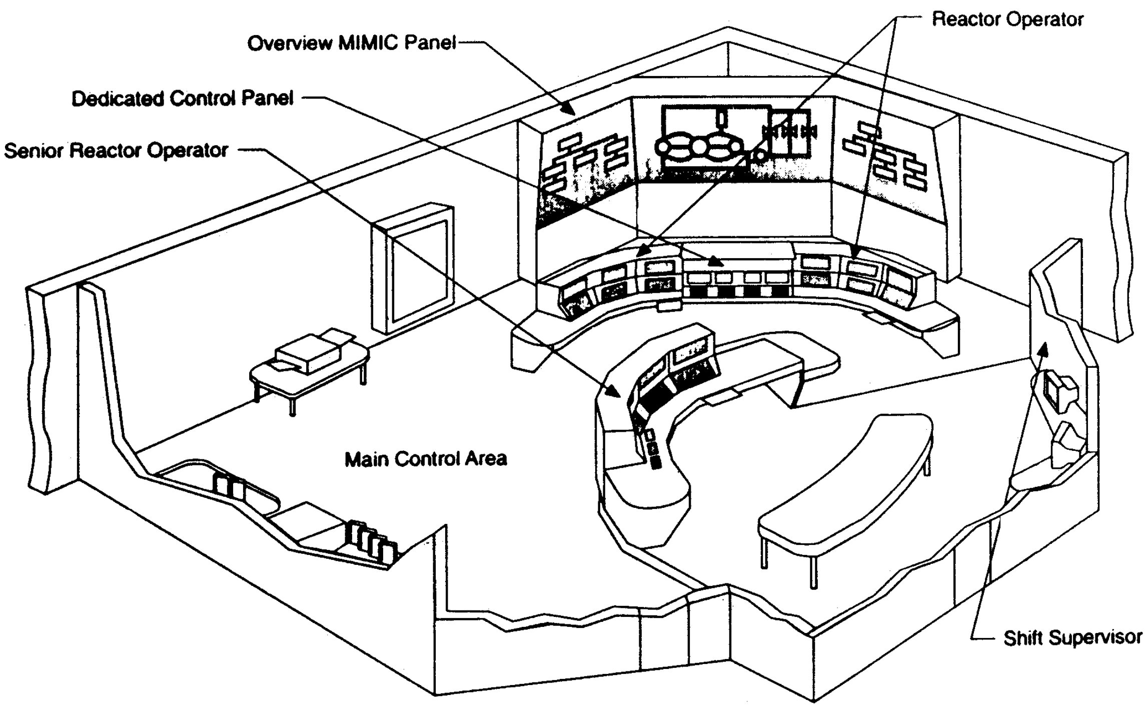


FIG. 4.10-4. AP1000 Main control room

4.10.4.2 Reactor protection system and other safety systems

The AP1000 provides instrumentation and controls to sense accident situations and initiate engineered safety features. The occurrence of a limiting fault, such as a loss-of-coolant accident or a secondary system break, requires a reactor trip plus actuation of one or more of the engineered safety features. This combination of events prevents or mitigates damage to the core and reactor coolant system components, and provides containment integrity.

The protection and safety monitoring system (PMS) provides the safety-related functions necessary to shut down the plant, and to maintain the plant in a safe shutdown condition. The protection and safety monitoring system controls safety-related components in the plant that may be operated from the main control room or remote shutdown workstation.

4.10.5 Electrical systems

The AP1000 on-site power system includes the main AC power system and the DC power system. The main AC power is a non-Class 1E system. The DC power system consists of two independent systems, one Class 1E and one non-Class 1E. The on-site power system is designed to provide reliable electric power to the plant safety and non-safety equipment for normal plant operation, startup, normal shutdown, accident mitigation, and emergency shutdown.

The main generator is connected to the off-site power system via three single-phase main step-up transformers. The normal power source for the plant auxiliary AC loads is provided from the 24 kV isophase generator buses through the two unit auxiliary transformers of identical ratings. In the event of a loss of the main generator, the power is maintained without interruption from the preferred power supply by an auto-trip of the main generator breaker. Power then flows from the main transformer to the auxiliary loads through the unit auxiliary transformers.

Off-site power has no safety-related function due to the passive safety features incorporated in the AP1000 design. Therefore, redundant off-site power supplies are not required. The design provides a reliable offsite power system that minimizes challenges to the passive safety system.

4.10.5.1 Operational power supply systems

The main AC power system is a non-Class 1E system that does not perform any safety functions. The standby power supply is included in the on-site standby power system.

The power to the main AC power system normally comes from the station main generator through unit auxiliary transformers. The plant is designed to sustain a load rejection from 100 percent power with the turbine generator continuing stable operation while supplying the plant house loads. The load rejection feature does not perform any safety function.

The on-site standby AC power system is powered by the two on-site standby diesel generators and supplies power to selected loads in the event of loss of normal, and preferred AC power supplies.

The plant DC power system comprises two independent Class 1E and non-Class 1E DC power systems. Each system consists of ungrounded stationary batteries, DC distribution equipment, and uninterruptible power supplies.

4.10.5.2 Safety-related systems

The Class 1E DC power system includes four independent divisions of battery systems. Any three of the four divisions can shut down the plant safely and maintain it in a safe shutdown condition. Divisions B and C have two battery banks. One of these battery banks is sized to supply power to selected safety-related loads for at least 24 hours, and the other battery bank is sized to supply power to another smaller set of selected safety-related loads for at least 72 hours following a design basis event (including the loss of all AC power).

For supplying power during the post-72 hour period following a design basis accident, provisions are made to connect an ancillary ac generator to the Class 1E voltage regulating transformers (Divisions B and C only). This powers the Class 1E post-accident monitoring systems, the lighting in the main control room, and ventilation in the main control room and Divisions B and C instrumentation and control rooms.

4.10.6 Safety concept

4.10.6.1 Safety requirements and design philosophy

The AP1000 design provides for multiple levels of defense for accident mitigation (defense-in-depth), resulting in extremely low core damage probabilities while minimizing the occurrences of containment flooding, pressurization, and heat-up. Defense-in-depth is integral to the AP1000 design, with a multitude of individual plant features capable of providing some degree of defense of plant safety. Six aspects of the AP1000 design contribute to defense-in-depth:

Stable Operation. In normal operation, the most fundamental level of defense-in-depth ensures that the plant can be operated stably and reliably. This is achieved by the selection of materials, by quality assurance during design and construction, by well-trained operators, and by an advanced control system and plant design that provide substantial margins for plant operation before approaching safety limits.

Physical Plant Boundaries. One of the most recognizable aspects of defense-in-depth is the protection of public safety through the physical plant boundaries. Releases of radiation are directly prevented by the fuel cladding, the reactor pressure boundary, and the containment pressure boundary.

Passive Safety-Related Systems. The AP1000 safety-related passive systems and equipment are sufficient to automatically establish and maintain core cooling and containment integrity for an indefinite period of time following design basis events assuming the most limiting single failure, no operator action and no onsite and offsite ac power sources.

Diversity within the Safety-Related Systems. An additional level of defense is provided through the diverse mitigation functions within the passive safety-related systems. This diversity exists, for example, in the residual heat removal function. The PRHR HX is the passive safety-related feature for removing decay heat during a transient. In case of multiple failures in the PRHR HX, defense-in-depth is provided by the passive safety injection and automatic depressurization (passive feed and bleed) functions of the passive core cooling system.

Non-safety Systems. The next level of defense-in-depth is the availability of certain non-safety systems for reducing the potential for events leading to core damage. For more probable events, these highly reliable non-safety systems automatically actuate to provide a first level of defense to reduce the likelihood of unnecessary actuation and operation of the safety-related systems.

Containing Core Damage. The AP1000 design provides the operators with the ability to drain the IRWST water into the reactor cavity in the event that the core has uncovered and is melting. This prevents reactor vessel failure and subsequent relocation of molten core debris into the containment. Retention of the debris in the vessel significantly reduces the uncertainty in the assessment of containment failure and radioactive release to the environment due to ex-vessel severe accident phenomena. (See Section 4.10.6.3 for additional discussion regarding in-vessel retention.)

AP1000 defense-in-depth features enhance safety such that no severe release of fission products is predicted to occur from an initially intact containment for more than 100 hours after the onset of core damage, assuming no actions for recovery. This time provides for performing accident management actions to mitigate the accident and prevent containment failure. The frequency of severe release as predicted by PRA is 1.95 x 10^{-8} per reactor year, which is much lower than for conventional plants.

4.10.6.2 Safety systems and features (active, passive, and inherent)

The AP1000 uses passive safety systems to improve the safety of the plant and to satisfy safety criteria of regulatory authorities. The use of passive safety systems provides superiority over conventional plant designs through 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 machinery 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 safety positions upon loss of power or upon receipt of a safety actuation signal. They are supported by multiple, reliable power sources to avoid unnecessary actuations.

The passive safety systems do not require the large network of active safety support systems (ac power, HVAC, cooling water, and the associated seismic buildings to house these components) that are needed in typical nuclear plants. As a result, support systems no longer must be safety class, and they are simplified or eliminated.

The AP1000 passive safety-related systems include:

- The passive core cooling system (PXS)
- The passive containment cooling system (PCS)
- The main control room emergency habitability system (VES)
- Containment isolation

These passive safety systems provide a major enhancement in plant safety and investment protection as compared with conventional plants. They establish and maintain core cooling and containment integrity indefinitely, with no operator or ac power support requirements. The passive systems are designed to meet the single-failure criteria, and PRAs are used to verify their reliability.

The AP1000 passive safety systems are significantly simpler than typical PWR safety systems since they contain significantly fewer components, reducing the required tests, inspections, and maintenance. They require no active support systems, and their readiness is easily monitored.

Emergency core cooling system- The passive core cooling system (PXS) (Figure 4.10-5) 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 US NRC-approved codes) demonstrate the effectiveness of the PXS in protecting the core following various RCS break events, even for breaks as severe as the 8-inch (200 mm) vessel injection lines. The PXS provides approximately a 76°F (42.2°C) 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 uses three passive sources of water to maintain core cooling through safety injection. These injection sources include the core makeup tanks (CMTs), the accumulators, and the IRWST. These injection sources are directly connected to two nozzles on the reactor vessel so that no injection flow can be spilled for the main reactor coolant pipe break cases.

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 squib valves. The tank is designed for atmospheric pressure, and therefore, the RCS must be depressurized before injection can occur.

The depressurization of the RCS is automatically controlled to reduce pressure to about 12 psig (0.18 MPa) which allows IRWST injection. The PXS provides for depressurization using the four stages of the ADS to permit a relatively slow, controlled RCS pressure reduction.

Passive residual heat removal - The PXS includes a 100% capacity passive residual heat removal heat exchanger (PRHR HX), which is connected through inlet and outlet lines to RCS loop 1. The PRHR HX protects the plant against transients that upset the normal steam generator feedwater and steam systems. The PRHR HX satisfies the safety criteria for loss of feedwater, feedwater and steam line breaks.

The IRWST provides the heat sink for the PRHR HX. 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. This steam condenses on the steel containment vessel and, after collection, drains by gravity back into the IRWST. The PRHR HX and the passive containment cooling system provide indefinite decay heat removal capability with no operator action required.

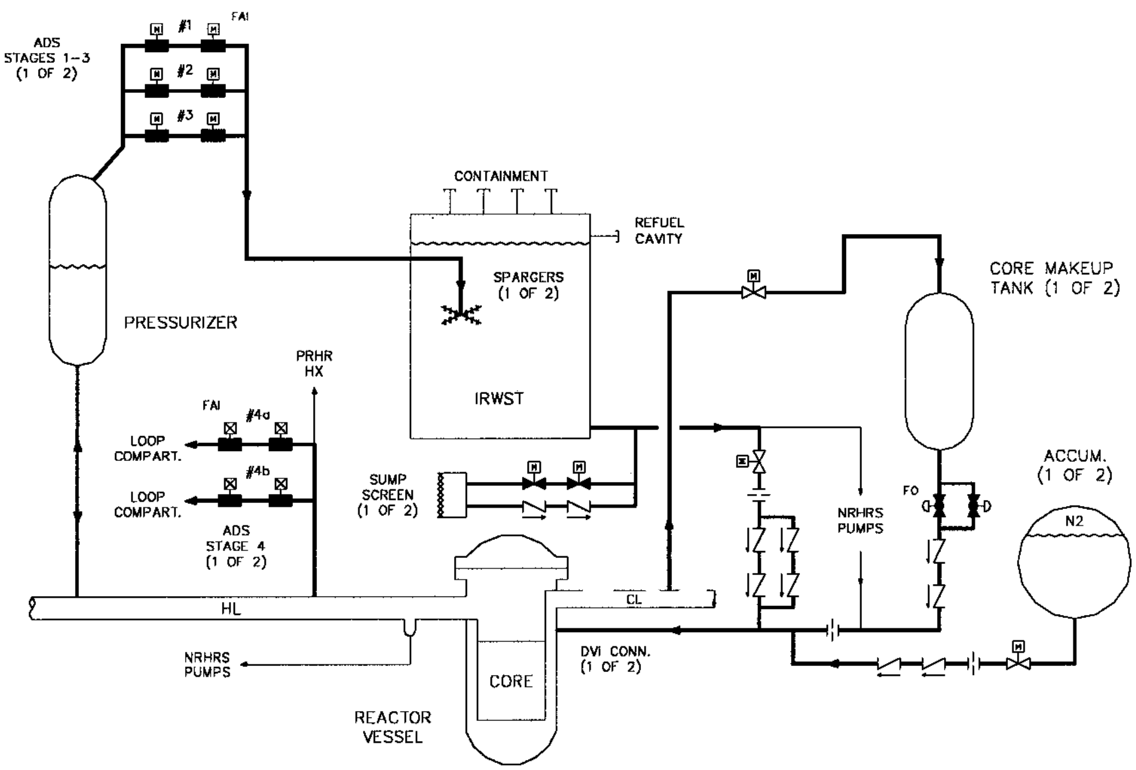


FIG. 4.10-5. AP1000 Passive core cooling system

Passive containment cooling - The passive containment cooling system (PCS) (Figure 4.10-6) 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 pressure is rapidly reduced and the design pressure is not exceeded.

The steel containment vessel 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 continuous natural circulation flow of air. During an accident, the air cooling is supplemented by evaporation of water whichdrains by gravity from a tank on top of the containment shield building.

Calculations have shown the AP1000 to have a significantly reduced large release frequency following a severe accident core damage scenario. With only the normal PCS air cooling, the containment stays well below the predicted failure pressure for at least 24 hours. Other factors include improved containment isolation and reduced potential for LOCAs outside of containment. This improved containment performance supports the technical basis for simplification of offsite emergency planning.

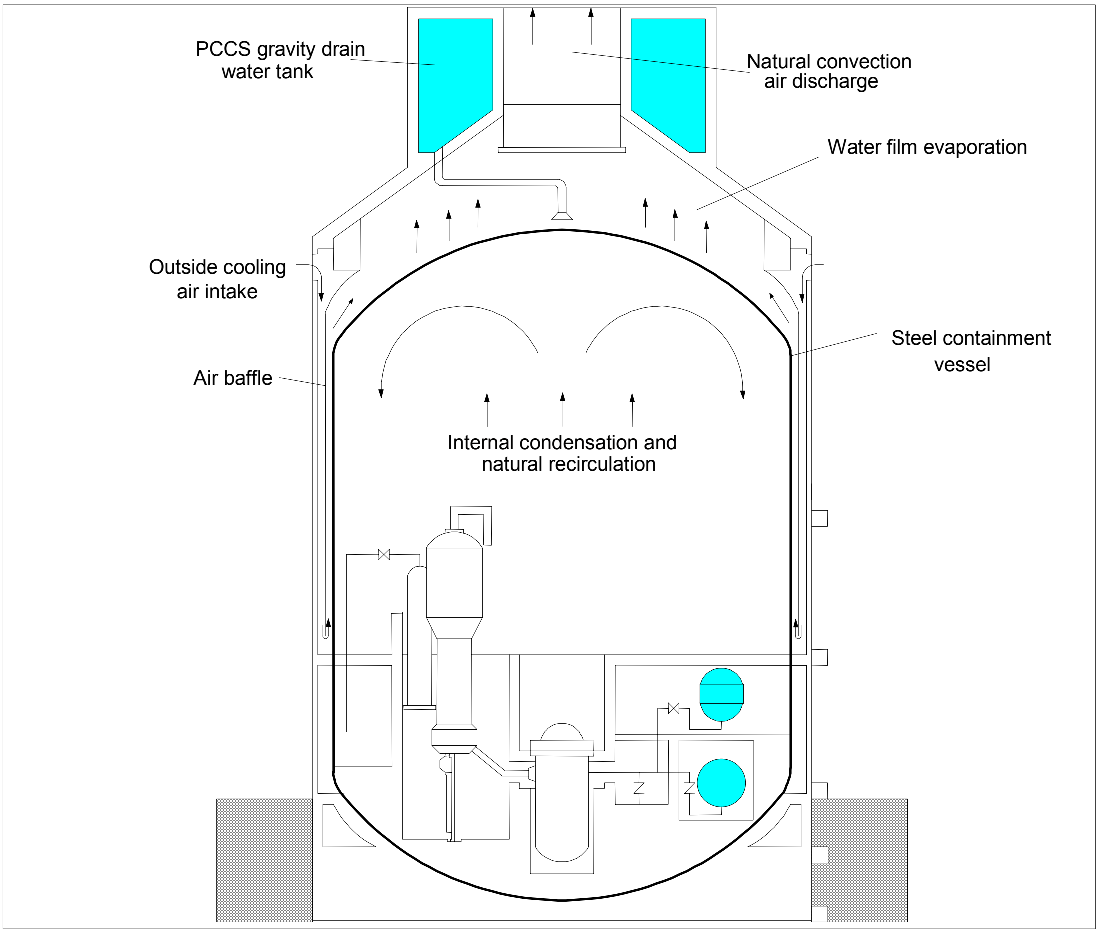


FIG. 4.10-6 AP1000 Passive containment cooling system

Main control room emergency habitability - The main control room emergency habitability system (VES) provides fresh air, cooling, and pressurization to the main control room (MCR) following a plant accident. Operation of the VES is automatically initiated upon receipt of a high MCR radiation signal, which isolates the normal control room ventilation path and initiates pressurization. Following system actuation, all functions are completely passive. The VES air supply is contained in a set of compressed air storage tanks. The VES also maintains the MCR at a slight positive pressure, to minimize the infiltration of airborne contaminants from the surrounding areas.

Containment isolation - AP1000 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. There are no penetrations required to support post-accident mitigation functions (the canned motor reactor coolant pumps do not require seal injection, and the passive residual heat removal and passive safety injection features are located entirely inside containment).

Long-term accident mitigation - A major safety advantage of the AP1000 versus current-day PWRs is that long-term accident mitigation is maintained by the passive safety systems without operator action and without reliance on offsite or onsite ac power sources. For the limiting design basis accidents, the core coolant inventory in the containment for recirculation cooling and boration of the core is sufficient to last for at least 30 days, even if inventory is lost at the design basis containment leak rate.

4.10.6.3 Severe accidents (beyond design basis accidents)

In-vessel retention of molten core debris - In-vessel retention (IVR) of molten core debris via water cooling of the external surface of the reactor vessel is an inherent severe accident management feature of the AP1000 passive plant. During postulated severe accidents, the accident management strategy to flood the reactor cavity with IRWST water and submerge the reactor vessel is credited with preventing vessel failure in the AP1000 PRA. The water cools the external surface of the vessel and prevents molten debris in the lower head from failing the vessel wall and relocating into the containment. Retaining the debris in the reactor vessel protects the containment integrity by preventing ex-vessel severe accident phenomena, such as ex-vessel steam explosion and core-concrete interaction, which have large uncertainties with respect to containment integrity.

The passive plant is uniquely suited to in-vessel retention because it contains features that promote external cooling of the reactor vessel. Figure 4.10-7 provides a schematic of the AP1000 reactor vessel, vessel cavity, vessel insulation and vents configuration that promotes IVR.

- The reliable multi-stage reactor coolant system (RCS) depressurization system results in low stresses on the vessel wall after the pressure is reduced.
- The vessel lower head has no vessel penetrations to provide a failure mode for the vessel other than creep failure of the wall itself.
- The reactor cavity can be flooded to submerge the vessel above the coolant loop elevation with water intentionally drained from the in-containment refueling water storage tank.
- The reactor vessel insulation design concept provides an engineered pathway for water-cooling the vessel and for venting steam from the reactor cavity.

The results of the AP1000 IVR analysis show that, with the AP1000 insulation designed to increase the cooling limitation at the lower head surface and the cavity adequately flooded, the AP1000 provides significant margin-to-failure for IVR via external reactor vessel cooling.

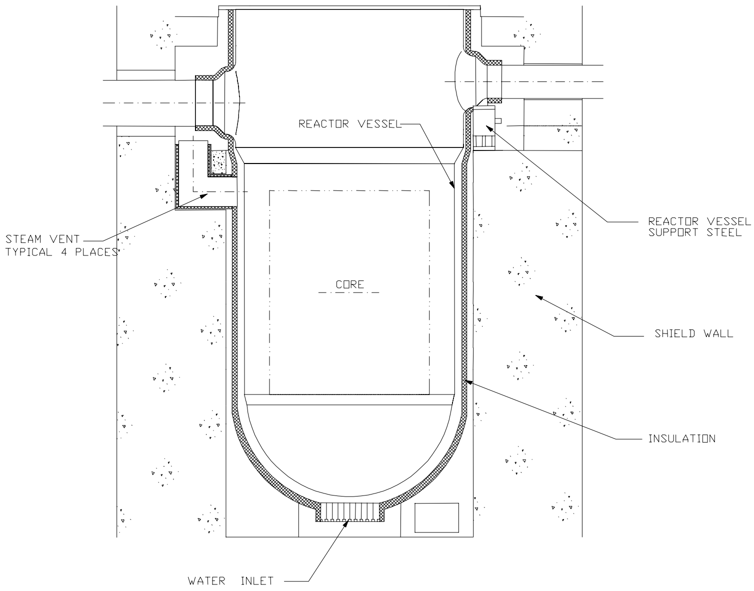


FIG. 4.10-7. AP1000 Configuration to Promote IVR of Molten Core Debris

4.10.7 Plant layout

4.10.7.1 Buildings and structures, including plot plan

A typical site plan for a single unit AP1000 is shown on Figure 4.10-8. The power block complex consists of five 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 are constructed on individual basemats. The nuclear island consists of the containment building, the shield building, and the auxiliary building, all of which are constructed on a common basemat.

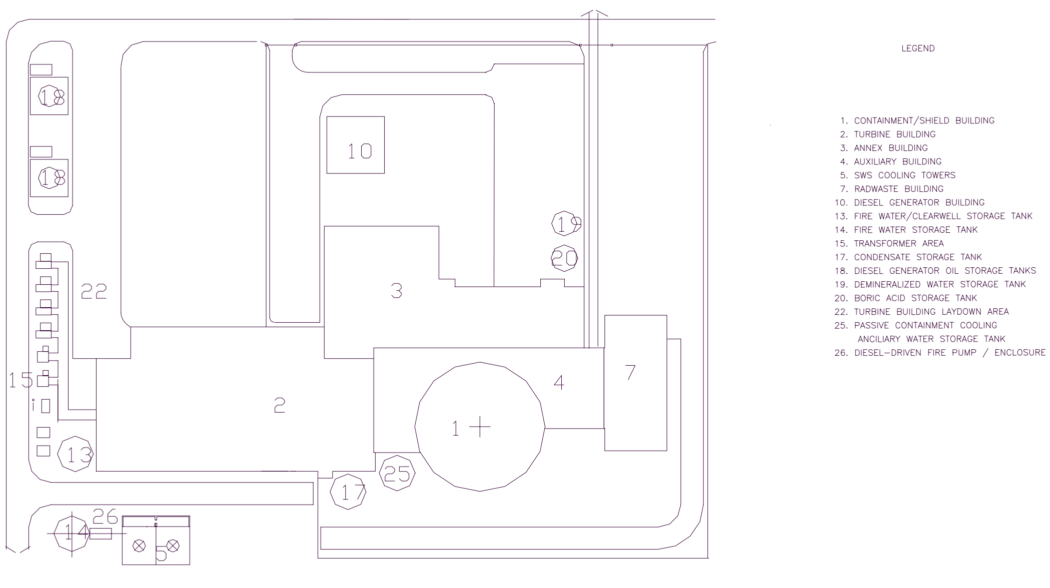


FIG. 4.10-8. AP1000 - Site layout

Plant arrangement - The AP1000 containment contains a 16-foot (4.9 m) diameter main equipment hatch and a personnel airlock at the operating deck level, and a 16-foot (4.9 m) diameter maintenance hatch and a personnel airlock at grade level. These large hatches significantly enhance accessibility to the containment during outages and, consequently, reduce the potential for congestion at the containment entrances. These containment hatches, located at the two different levels, allow activities occurring above the operating deck to be unaffected by activities occurring below the operating deck.

The containment arrangement provides significantly larger laydown areas than most conventional plants at both the operating deck level and the maintenance floor level. Additionally, the auxiliary building and the adjacent annex building provide large staging and laydown areas immediately outside of both large equipment hatches.

4.10.7.2 Reactor building

The reactor building of the AP1000 is a shield building surrounding the containment (Section 4.10.7.3).

4.10.7.3 Containment

Containment building - The containment building is the containment vessel and all structures contained within the containment vessel. The containment building is an integral part of the overall containment system with the functions of containing the release of airborne radioactivity following postulated design basis accidents and providing shielding for the reactor core and the reactor coolant system during normal operations.

The containment vessel 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.

The principal systems located within the containment building are the reactor coolant system, the passive core cooling system, and the reactor coolant purification portion of the chemical and volume control system.

Shield building - The shield building is the structure and annulus area that surrounds the containment vessel. During normal operations the shield building, in conjunction with the internal structures of the containment building, provides the required shielding for the reactor coolant system and all the other radioactive systems and components housed in the containment. During accident conditions, the shield building provides the required shielding for radioactive airborne materials that may be dispersed in the containment as well as radioactive particles in the water distributed throughout the containment.

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 passive containment cooling system air baffle is to provide a pathway for natural circulation of cooling air in the event that a design basis accident results in a large release of energy into the containment. In this event the outer surface of the containment vessel transfers heat to the air between the baffle and the containment shell. This heated and thus, lower density air flows up through the air baffle to the air diffuser and cooler and higher density air is drawn into the shield building through the air inlet in the upper part of the shield building.

Another function of the shield building is to protect the containment building from external events. The shield building protects the containment vessel and the reactor coolant system from the effects of tornadoes and tornado produced missiles.

4.10.7.4 Turbine building

The turbine building houses the main turbine, generator, and associated fluid and electrical systems. It provides weather protection for the laydown and maintenance of major turbine/generator components. The turbine building also houses the makeup water purification system. No safety-related equipment is located in the turbine building.

4.10.7.5 Other buildings

Auxiliary building - The primary function of the auxiliary building is to provide protection and separation for the safety-related seismic Category I mechanical and electrical equipment located outside the containment building. The auxiliary building provides protection for the safety-related equipment against the consequences of either a postulated internal or external event. The auxiliary building also provides shielding for the radioactive equipment and piping that is housed within the building. The most significant equipment, systems contained within the auxiliary building are the main control room, I&C systems, electrical power systems, fuel handling area, mechanical equipment areas, containment penetration areas, and the main steam and feedwater valve compartments.

The primary function of the auxiliary building is to provide protection and separation for the safety-related seismic Category I mechanical and electrical equipment located outside the containment building. The auxiliary building provides protection for the safety-related equipment against the consequences of either a postulated internal or external event. The auxiliary building also provides shielding for the radioactive equipment and piping that is housed within the building.

The most significant equipment, systems, and functions contained within the auxiliary building are the following:

- Main control room
- Class 1E instrumentation and control systems
- Class 1E electrical system
- Fuel handling area
- Mechanical equipment areas
- Containment penetration areas
- Main steam and feedwater isolation valve compartment

Main Control Room – The main control room provides the human system interfaces required to operate the plant safely under normal conditions and to maintain it in a safe condition under accident conditions. The main control room includes the main control area, the operations staff area, the switching and tagging room and offices for the shift supervisor and administrative support personnel.
Instrumentation and Control Systems – The protection and safety monitoring system and the plant control system provide monitoring and control of the plant during startup, ascent to power, powered operation, and shutdown. The instrumentation and control systems include the protection and safety monitoring system, the plant control system, and the data display and processing system.

Class 1E Electrical System – The Class 1E system provides 125 volts dc power for safety-related and vital control instrumentation loads including monitoring and control room emergency lighting. It is required for safe shutdown of the plant during a loss of ac power and during a design basis accident with or without concurrent loss of offsite power.

Fuel Handling Area – The primary function of the fuel handling area is to provide for the handling and storage of new and spent fuel. The fuel handling area in conjunction with the annex building provides the means for receiving, inspecting and storing the new fuel assemblies. It also provides for safe storage of spent fuel as described in the DCD. The fuel handling area provides for transferring new fuel assemblies from the new fuel storage area to the containment building and for transferring spent fuel assemblies from the containment building to the spent fuel storage pit within the auxiliary building.

The fuel handling area provides for removing the spent fuel assemblies from the spent fuel storage pit and loading the assemblies into a shipping cask for transfer from the facility. This area is protected from external events such as tornadoes and tornado produced missiles. Protection is provided for the spent fuel assemblies, the new fuel assemblies and the associated radioactive systems from external events. The fuel handling area is constructed so that the release of airborne radiation following any postulated design basis accident that could result in damage to the fuel assemblies or associated radioactive systems does not result in unacceptable site boundary radiation levels.

Mechanical Equipment Areas – The mechanical equipment located in radiological control areas of the auxiliary building are the normal residual heat removal pumps and heat exchangers, the spent fuel cooling system pumps and heat exchangers, the solid, liquid, and gaseous radwaste pumps, tanks, demineralizers and filters, the chemical and volume control pumps, and the heating, ventilating and air conditioning exhaust fans.

The mechanical equipment located in the clean areas of the auxiliary building are the heating, ventilating and air conditioning air handling units, associated equipment that service the main control room, instrumentation and control cabinet rooms, the battery rooms, the passive containment cooling system recirculation pumps and heating unit and the equipment associated with the air cooled chillers that are an integral part of the chilled water system.

Containment Penetration Areas – The auxiliary building contains all of the containment penetration areas for mechanical, electrical, and instrumentation and control penetrations. The auxiliary building provides separation of the radioactive piping penetration areas from the non-radioactive penetration areas and separation of the electrical and instrumentation and control penetration areas from the mechanical penetration areas. Also provided is separation of redundant divisions of instrumentation and control and electrical equipment.

Main Steam and Feedwater Isolation Valve Compartment – The main steam and feedwater isolation valve compartment is contained within the auxiliary building. The auxiliary building provides an adequate venting area for the main steam and feedwater isolation valve compartment in the event of a postulated leak in either a main steam line or feedwater line.

Annex building - The annex building provides the main personnel entrance to the power generation complex. It includes accessways for personnel and equipment to the clean areas of the nuclear island in the auxiliary building and to the radiological control area. The building includes the health physics facilities for the control of entry to and exit from the radiological control area as well as personnel support facilities such as locker rooms. The building also contains the non-1E ac and dc electric power systems, the ancillary diesel generators and their fuel supply, other electrical equipment, the technical support center, and various heating, ventilating and air conditioning systems. No safety-related equipment is located in the annex building.

The annex building includes the health physics facilities and provides personnel and equipment accessways to and from the containment building and the rest of the radiological control area via the auxiliary building. Provided are large, direct accessways to the upper and lower equipment hatches of the containment building for personnel access during outages and for large equipment entry and exit. The building includes a hot machine shop for servicing radiological control area equipment. The hot machine shop includes decontamination facilities including a portable decontamination system that may be used for decontamination operations throughout the nuclear island.

Diesel generator building - The diesel generator building houses two identical slide along diesel generators separated by a three-hour fire wall. These generators provide backup power for plant operation in the event of disruption of normal power sources. No safety-related equipment is located in the diesel generator building.

Radwaste building - The radwaste building includes facilities for segregated storage of various categories of waste prior to processing, for processing by mobile systems, and for storing processed waste in shipping and disposal containers. No safety-related equipment is located in the radwaste building. Dedicated floor areas and trailer parking space for mobile processing systems is provided for the following:

- Contaminated laundry shipping for offsite processing
- Dry waste processing and packaging
- Hazardous/mixed waste shipping for offsite processing
- Chemical waste treatment
- Empty waste container receiving and storage
- Storage and loading packaged wastes for shipment.

The radwaste building also provides for temporary storage of other categories of plant wastes.

4.10.8 Technical data

General plant data

Power plant output, gross	1200	MWe
Power plant output, net	1117	MWe
Reactor thermal output [core power 3400 MWt]	3415	MWt
Power plant efficiency, net	33	%
Cooling water temperature	30.5	°C

Nuclear steam supply system

Number of coolant loops	2 hot legs/4 cold legs	
Steam flow rate at nominal conditions	1886	kg/s
Feedwater flow rate at nominal conditions	1887	kg/s
Steam temperature/pressure	272.9/5.76	°C/MPa
Feedwater temperature	226.7	°C

Reactor coolant system

Primary coolant flow rate, per loop	9.94	m^3/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	280.7	°C
Coolant outlet temperature, at RPV outlet	321.1	°C
Mean temperature rise across core	40.4	°C

Reactor core

Active core height	4.267	m
Equivalent core diameter	3.04	m
Heat transfer surface in the core	5268	m^2
Fuel inventory		84.5 t U
Average linear heat rate	18.7	kW/m
Average fuel power density	40.2	kW/kg U
Average core power density (volumetric)	109.7	kW/l
Thermal heat flux, F_q	2.60	kW/m^2
Enthalpy rise, F_H	1.65	
Fuel material	Sintered UO_2	
Fuel assembly total length	4 795	mm
Rod array	square, 17×17 (XL)	
Number of fuel assemblies	157	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	24	
Number of structural spacer grids	10	
Number of intermediate flow mixing grids	4	
Enrichment (range) of first core	2.35-4.45	Wt% U-235
Enrichment of reload fuel at equilibrium core	4.8	Wt% U-235
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel (nominal)	60 000	MWd/t
Cladding tube material	ZIRLO ™	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly	799.7	kg
Burnable absorber, strategy/material	Discrete burnable absorber, Integral fuel burnable absorber	
Number of control rods	69 (53 black, 16 gray)	
Absorber rods per control assembly	24	
Absorber material	Ag-In-Cd (black), Ag-In-Cd/304SS (gray)	
Drive mechanism	Magnetic jack	
Positioning rate [in steps/min or mm/s]	45	steps/min
Soluble neutron absorber	Boric acid	

Reactor pressure vessel

Cylindrical shell inner diameter	3 988	mm
Wall thickness of cylindrical shell	203	mm
Total height	12056	mm
Base material: cylindrical shell	Carbon steel	
RPV head	Carbon steel	
Liner	Stainless steel	
Design pressure/temperature	17.1/ 343.3	MPa/°C

Steam generators

Type	Delta 125, vertical, U-tube	
Number	2	
Heat transfer surface	11477	m^2
Number of heat exchanger tubes	10025	

Tube dimensions	17.5/15.4	mm
Maximum outer diameter	5575.3	mm
Total height	22460	mm
Transport weight	663.7	t
Shell and tube sheet material	Carbon steel	
Tube material	Inconel 690-TT	

<u>Reactor coolant pump</u>

Type	Canned motor	
Number		4
Design pressure/temperature	17.1/343.3	MPa/^{o}C
Rated flow rate	4.97	m^3/s
Rated head	111.3	m
Pump speed (nominal)	1750	rpm

<u>Pressurizer</u>

Total volume	59.47	m^3
Steam volume: nominal full load	31.14	m^3
Design pressure/temperature	17.1/360	MPa/^{o}C
Heating power of the heater rods	1600	kW
Inner diameter	2.28	m
Total height (surge nozzle safe end to spray nozzle safe end)	16.27	m

<u>Pressurizer relief tank</u> Not applicable

<u>Primary containment</u>

Type	Dry, free standing, steel	
Overall form (spherical/cyl.)	Cylindrical	
Dimensions (diameter/height)	39.6/65.63	m
Design pressure/temperature (DBEs)	406.7/148.9	kPa-g/^{o}C
(severe accident situations)	889.4 /204.4	kPa-g/^{o}C
Design leakage rate	0.10	vol%/day
Is secondary containment provided?	Only around containment penetration area	
Material	SA 738, Grade B	

<u>Safety injection</u>

Passive residual heat removal		
Number heat exchangers	1	
Type	Vertical C-tube	
Heat transfer,design	$2.01x10^8$	Btu/hr
Design pressure/temperature	17.2/343.3	MPa/ ^{o}C
Core Makeup Tanks		
Number		2
Volume	70.8	m^3
Design pressure/temperature	17.2/343.3	MPa/ ^{o}C
Accumulators		
Number	2	
Volume	56.6	m^3
Design pressure/temperature	5.6/148.9	MPa/ ^{o}C
In-containment Refueling Water Storage Tank (IRWST)		
Number	1	
Volume (minimum)	2092.6	m^3
Design pressure/temperature	0.14/65.6	MPa/ ^{o}C

<u>Reactor auxiliary systems</u>

Reactor water cleanup, capacity	6.3	kg/s
(chemical volume & control) filter type	Cartridge	
Residual heat removal, shutdown cooling	89.3	kg/s
(normal RHR)	low pressure makeup	68.9 kg/s

<u>Power supply systems</u>

Main transformer, rated voltage	24 kV/site specific	
rated capacity	1250	MVA
Unit auxiliary transformers, rated voltage	24/6.9	kV
rated capacity	70	MVA
Start-up transformer, rated voltage	Site specific/6.9 kV	
rated capacity	70	MVA
Medium voltage busbars	6	
Number of low voltage busbar systems	10	
Standby diesel generating units: number	2	
rated power	4	MW
Number of diesel-backed busbar systems	2	

Voltage level of these	6900	V ac
Number of DC distributions	10	
Voltage level of these	125	V dc
Number of battery-backed busbar systems	11	
Voltage level of these	125	V ac

<u>Turbine plant</u>

Number of turbines per reactor	1	
Type of turbine(s)	Tandem-compound, 6-flow, 54 in. (1372 mm) last-stage blade	
Number of turbine sections per unit	1HP/ 3LP	
Turbine speed	1800 rpm 1500	rpm (for 50 Hz)
HP inlet pressure/temperature	5.5/271	MPa/^{o}C

<u>Generator</u>

Type	3-phase, synchronous	
Rated power	1250	MVA
Active power	1200	MWe
Voltage	24	kV
Frequency	50 / 60	Hz

<u>Condenser</u>

Type	Multipressure, single pass	
Cooling water flow rate	37.85	m^3/s
Cooling water temperature	30.5	^{o}C
Condenser pressure	9.1	kPa

<u>Condensate clean-up system</u>

Full flow/part flow	part flow, 33%	
Filter type	Deep bed	

<u>Feedwater pumps</u>

Main Feedwater Pumps		
Number	3	
Startup Feedwater Pumps		
Number	2	
Design flow rate	0.033	m^3/s
Pump head	990.6	m
Feedwater temperature	26.7	
Pump Speed	3600	rpm

<u>Condensate and feedwater heaters</u>

Number of heating stages	6

4.10.9 Measures to enhance economy and maintainability

The AP1000 is a logical extension of the AP600 design. The AP1000 maintains the same design philosophy of AP600, such as use of proven components, systems simplification and state-of-the-art construction techniques. The AP1000 optimizes the power output while maintaining the AP600 NI footprint, to reduce capital and generation costs.

Simplification - AP1000 is an advance passive plant that has been designed to meet globally recognized requirements. A concerted effort has been made to simplify systems and components, to facilitate construction, operation and maintenance and to reduce the capital and generating costs.

The use of passive systems allows the plant design to be significantly simpler. In addition, the passive safety systems do not require the large network of safety support systems found in current generation nuclear power plants (e.g., Class 1E ac power, safety HVAC, safety cooling water systems and associated seismic buildings). The AP1000 uses 50% fewer valves, 83% less pipe (safety grade), 87% less cable, 36% fewer pumps, and 56% less seismic building volumes.

Simplicity reduces the cost for reasons other than reduction of the number of items to be purchased. With a fewer number of components, installation costs are reduced, construction time is shortened and maintenance activities are minimized.

Construction Schedule- The AP1000 has been designed to make use of modern modular construction techniques. Not only does the design incorporate vendor designed skids and equipment packages, it also includes large structural modules and special equipment modules. Modularization allows construction tasks that were traditionally performed in sequence to be completed in parallel. The modules, constructed in factories, can be assembled at the site for a planned construction schedule of 3 years – from ground-breaking to fuel load. This duration has been verified by experienced construction managers through 4D (3D models plus time) reviews of the construction sequence.

Availability and O&M Costs - The AP1000 combines proven Westinghouse PWR technology with utility operating experience to enhance reliability and operability. Steam generators are similar to the recent replacement steam generators, and canned motor pumps and rugged turbine generators are proven performers with outstanding operating records. The Digital on-line diagnostic instrumentation and control system features an integrated control system that avoids reactor trips due to single channel failure. In addition, the plant design provides large margins for plant operation before reaching the safety limits. This assures a stable and reliable plant operation with a reduced number of reactor trips (less than one per year). Based on the above, and considering the short planned refueling outage (17 days) and plans to use a 18 to 24-month fuel cycle, the AP1000 is expected to exceed the 93% availability goal.

For AP1000 availability is enhanced by the simplicity designed into the plant, as described above. There are fewer components which result in lower maintenance costs, both planned and unplanned. In addition, the great reduction in safety-related components results in a large reduction in inspection and tests. Simplicity is also reflected in the reduced AP1000 staffing requirements.

4.10.10 Project status and planned schedule

The AP1000 is based extensively on the AP600 passive plant design that received Final Design Approval and Design Certification from the United States Nuclear Regulatory Commission (US NRC) in September 1998 and December 1999, respectively. Westinghouse and the U.S. NRC have completed a 12-month pre-licensing review of the AP1000 that established the applicability of AP600 tests and selected safety analysis computer codes to the AP1000 design certification application. On March 28, 2002, following the pre-application review, Westinghouse submitted an Application for Final Design Approval and Design Certification for the AP1000. The U.S. NRC completed its Review for Acceptance and docketed the Application on June 25 2002. The docketing of the application signifies that its content is acceptable for review by the U.S. NRC as a complete safety case, in accordance with appropriate U.S. regulations. The U.S. NRC issued their Draft Safety Evaluation Report for AP1000 on June 16, 2003. The U.S. NRC expects to complete the review of the AP1000 and issue a Final Design Approval (FDA) in 2004.

4.11 EP-1000 (WESTINGHOUSE, USA/ANSALDO, ITALY)

4.11.1 Introduction

In 1994, a group of European utilities initiated, together with Westinghouse and its industrial partner ANSALDO, a program designated EPP (European Passive Plant) to evaluate Westinghouse passive nuclear plant technology for application in Europe.

Phase 1 of the European passive plant program, completed in 1996, involved evaluation of the Westinghouse 600 MW(e) AP600 and 1000 MW(e) simplified pressurized water reactor (SPWR) designs against the European Utility Requirements (EUR), and when necessary, investigation of possible modifications to achieve compliance with the EUR.

Phase 2A of the EPP program started in 1997 and was concluded in 1998. The program developed a 1000 MW(e), three loop, reference passive plant design (EP1000). With respect to safety systems and containment, the EP1000 three loop reference plant design closely followed that of the Westinghouse SPWR nuclear island (NI) layout design, while the AP600 plant design was taken as the basis for the EP1000 reference design in the auxiliary system design areas. Phase 2A of the program focused on improving the design of important systems and structures with the key design goal of meeting the EUR.

Phase 2B of the EPP program (1999–2001) was called the Design Verification Phase. This phase of the program focused on the verification of the EP1000 plant design and cost assessment. Significant work was performed in the areas of plant safety assessment for the preparation of a “Safety Case Report”, plant design verification via formalized Design Review activities, and development of a European Market Cost Assessment.

In 1999, in parallel with the Phase 2B activities, Westinghouse, in response to U.S. market conditions, which indicated that new nuclear units must be competitive with natural gas fired generation alternatives, initiated a study to evaluate the feasibility of uprating the AP600 (2-loop passive plant design) to achieve better plant economics needed to compete in the U.S. In order to develop a cost competitive nuclear power plant, Westinghouse has undertaken a program to develop and license a larger version of the AP600 with an increased power output of greater than 1000 MW(e) (3400 MW(t)), while maintaining the AP600 design configuration, use of proven components and licensing basis. The plant is called AP1000.

The approach to achieving these objectives is to design the AP1000 within the space constraints of the AP600, while retaining the credibility of proven components and substantial safety margins. The arrangement of the reactor, the passive safety systems and the auxiliary systems is the same as the AP600. To increase the power output of the reactor, the core, reactor coolant pumps and steam generators have been increased in size. The design of these larger reactor components are based on components that are used in operating PWRs or have been developed / tested for new PWRs. In order to maintain adequate safety margins, the capacity of the passive safety features have been selectively increased based on insights from the AP600 test and analysis results. With this approach a major portion of the AP600 Design Certification Document (DCD) directly applies to AP1000.

The EPP utilities decided, before the end of EPP Phase 2B, to adopt the AP1000 as the new EPP reference plant design. Adopting the AP1000 as the EPP reference design provides a more cost effective way to achieve the objective of the EPP Agreement; i.e., a 1000 MW(e) PWR design based on passive technology that meets the EUR and is licensable in Europe. The additional design and licensing detail, available through the AP1000 program, provides significant cost and schedule advantages to implement the first passive PWR plant(s) in Europe, once a decision to build is made. Since the design activities focus on a single passive PWR design intended to meet new plant requirements in Europe, America and Asia, this will provide greater opportunity to realize the economic advantages of a standard plant built in multiple locations, since specialized designs for specific regions require heavier investment. The ongoing EPP Program is following this approach.

The AP1000 has implemented some design improvements developed during the EPP program, including low boron core capabilities. The following sections discuss design areas where the AP1000 U.S. design will be assessed according to the EUR for its application in Europe. The remaining design features are described in the section of this document discussing the AP1000 design.

4.11.2 Description of the nuclear systems

Refer to this section in the AP1000 design description for this information.

4.11.2.1 Primary circuit and its main characteristics

Refer to this section in the AP1000 design description for this information.

4.11.2.2 Reactor core and fuel design

4.11.2.2.1 Core Design (MOX Fuel/Low Boron Core)

The EUR policy is to have sufficient flexibility in the core design to be able to optimize both the fuel cycle costs (which include back end costs) and the operation and maintenance costs against conditions which may change over the plant lifetime.

The EUR (requirement EUR 2.2.3.1) requires that the EP1000 core shall be optimized for UO_2 fuel assemblies. However, provisions shall be made to allow the use of up to 50% standard mixed U-Pu oxide (MOX) fuel assemblies in the core, the remaining being UO_2 assemblies.

According to the EUR, the core shall allow flexibility of operating for fuel cycles with refueling intervals from 12 to 24 months (requirement EUR 2.2.3.2), assuming an 87% capacity factor (requirement EUR 2.2.7.2.1).

The core should also be capable of operating with low boron concentration (requirement EUR 2.2.3.7). The low boron capability sets out to reduce boron dilution risks and to produce an adequately negative moderator feedback to avoid damage of the core and RCS pressure boundary during any anticipated transient without scram (ATWS) event through 100% core life (requierment EUR 2.8.1.1.3), and finally reduces ALARA costs through improvement of the coolant chemistry.

4.11.2.2.1.1 Low Boron Core

The EPP Phase 2B detailed core design has shown the feasibility of a 24 month Low Boron core capable to meet all EUR requirements.

Benefits obtained both during operation (e.g., manuvering at beginning of life (BOL), ALARA considerations) and accidental conditions (e.g., Boron Dilution, ATWS) have suggested the implementation of a Low Boron Core design for the U.S. AP1000 plant.

A Low Boron Core design has been developed during the EPP Phase 2C program and has been adopted as the reference core design for the AP1000. Refer to the “Reactor core and fuel design” section of the AP1000 design description for the main characteristics of the AP1000 core design.

The low boron core design is a clear example of the close integration of the European project with the the U.S. program and confirms the important role of the European Passive Plant Program, and the positive influence of the EUR program, for the development of new passive plant designs.

4.11.2.2.1.2 MOX Core

A conceptual 24 month equilibrium cycle MOX core design, developed during the early stages of the EPP program, showed that the EP1000 core can load up to 50% MOX fuel and achieve an 18 month cycle length.

The EP1000 three loop 18-month, 50% MOX core design studies, performed during EPP Phase 2A, indicate that the MOX core design is capable of meeting the European requirements. It is expected, based on EPP Phase 2A activities, that a 50% MOX core for the AP1000 design will meet Europeam Requirements.

4.11.2.3 Fuel handling and transfer systems

The EUR specifies the spent fuel pool shall be designed to accommodate 15 years of MOX spent fuel plus 10 years of UO_2 spent fuel plus one full core offload. According to the EUR, the full core off load, for the evaluation of the Fuel Pool and Cleaning System design heat load, should take place starting at 100 hours after shutdown and be completed at 148 hours after shutdown. The associated cooling system shall be designed to limit the pool temperature to less than 60°C (140°F) at all times except during a full core offload where the temperature shall be less than 80°C (176°F).

The Spent Fuel Pool Cooling Water System (SFS) is affected by the higher decay heat of MOX fuel in the long term with respect to the standard UO_2 fuel (i.e., the MOX decay heat at 15 years can be up to three times that of UO_2). The Component Cooling Water System (CCS) must also remove a higher overall heat load, although the percentage increase is small. The reference AP1000 SFS and CCS heat exchanger size, pump size, and piping size will be assessed according to the EUR requirements.

4.11.2.4 Primary components

Refer to this section in the AP1000 design description for this information.

4.11.2.5 Reactor auxiliary systems

Refer to this section in the AP1000 design description for this information.

4.11.2.6 Operating characteristics

Refer to this section in the AP1000 design description for this information.

4.11.3. Description of the turbine generator plant systems

Refer to this section in the AP1000 design description for this information.

4.11.3.1 Turbine generator plant

Refer to this section in the AP1000 design description for this information.

4.11.3.2 Condensate and feedwater systems

Refer to this section in the AP1000 design description for this information.

4.11.3.3 Auxiliary Systems

The 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)

The function of the Component Cooling Water System, and Service Water System is to provide sufficient cooling to the auxiliary plant loads during normal plant operation including: Power Operation, Shutdown, Cooldown, Refueling and Plant Startup.

The RNS is designed to remove both residual and sensible heat from the core and the Reactor Coolant System (RCS) and to reduce the temperature of the RCS during the second phase of the plant cooldown, being that the first phase of the cooldown is accomplished by transferring heat from the RCS via the Steam Generators to the Main Steam System.

EUR criteria that impact the design of the heat removal systems include:

- Use of MOX fuel,
- Boron Recycling (waste evaporator heat loads),
- More rapid plant cooldown times,
- Site parameters (e.g., max/min ultimate heat sink temperatures),
- Increased spent fuel storage capability,
- Aggressive refueling schedule that requires full core off load at about 108 hours to be able to meet the EUR 17 day refueling requirement.

4.11.3.3.1 RNS and CCS Designs

The European Utility Requirements (use of MOX fuel, Boron Recycling, Cooldown time limits and site conditions) directly affect the design of the above Auxiliary Cooling Systems.

The Heat Removal Design Bases set by the EUR, that directly affect the design of RNS and CCS, 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 (requirement EUR 2.2-2.1);
- Initiation of system (RNS) operation 6 hrs after reactor shutdown (requirement EUR 2.8 - 2.4.1.3.1);
- Maintain RCS T < 60 °C (140 °F) during refueling with one train unavailable, beyond a time period from reactor shutdown compatible with availability targets (requirement EUR 2.8 - 2.4.1.3.1);
- Maintain RCS T at 180 °C (356 °F) during Hot Shutdown with one train available, within 12 hrs after shutdown (requirement EUR 2.8 - 2.4.1.3.1);
- The RNS shall have sufficient capacity to bring the reactor to 90 °C (194 °F) within 36 hrs after shutdown with a single failure in the RNS system (requirement EUR 2.8 - 2.4.1.3.1).

The reference AP1000 auxiliary systems will be assessed according to these EUR requirements.

4.11.3.3.2 Boron Recycle

The EUR requires that the boron utilized in the reactor plant be recycled (requirement EUR 2.8.1.2.4.1). This is a change from the AP1000 reference plant design (U.S. AP1000) which does not recycle boron. As a result of this EUR requirement, EPP studies have investigated the Chemical and Volume Control System (CVS) and Liquid Radwaste System (WLS) designs to accommodate boron recycling. The studies have shown that a boron recycle system can be implemented into the reference AP600/AP1000 designs.

4.11.4 Instrumentation and control systems

Refer to this section of the AP1000 design description for this information.

4.11.5 Electrical systems

EPP Phase 2C design activities included development of a 50 HZ Main AC Power System (ECS) for AP1000.

In particular, the following design activities have been performed:

- Establish electrical loads,
- Equipment Sizing,
- NI Layout confirmation,
- Develop ECS system specification document and one line diagram.

The results of the design activities demonstrate there is no impact on the overall AP1000 plant design to implement a 50 Hz electrical system.

4.11.6 Safety Concept

4.11.6.1 Safety requirements and design philosophy

Refer to this section in the AP1000 design description for this information.

4.11.6.2 Safety systems and features (active, passive and inherent)

Refer to this section in the AP1000 design description for this information.

4.11.6.3 Severe accidents (beyond design basis accidents)

The assessment of performance against severe accidents is performed in agreement with European Utilities Requirements. The general approach to severe accidents is to identify 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 Accidents (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 (DBCs).

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 DBCs.

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.

Analyses have been performed in the EPP program to assess the consequences of Complex Sequences such as Anticipated Transients Without Scram (ATWS) and Multiple Steam Generator Tube Ruptures

(MSGTR) for AP1000. Results have shown that the AP1000 can be expected to meet EUR severe accident criteria without modifications from the reference design licensed in the U.S.

Aircraft Crash Protection

The EUR includes requirements for considering the effects of an aircraft crash and gas cloud explosion on the plant. The EUR approach to airplane crash calls for site or country specific probabilistic assessment of the public safety risks associated with airplane crash at a nuclear power plant, and requires providing measures needed to keep this risk acceptably low.

Protection against aircraft crash loads, representative of a military fighter aircraft, has been studied during the ealry phases of the EPP program. Based on the results of these studies, for locations requiring protection for such loads, protection can be addressed in the AP1000 Nuclear Island structures by means of sufficiently thick/reinforced external walls and roofs, avoiding direct connection of safe shutdown components to external walls, and consideration of structural dynamic response in the selection and support of safe shutdown components.

4.11.7 Plant layout

Refer to this section in the AP1000 design description for this information.

4.11.8 Technical data

Refer to this section in the AP1000 design description for this information.

4.11.9 Measures to enhance economy and maintainability

Refer to this section in the AP1000 design description for this information.

4.11.10 Project status and planned schedule

The EPP program has adopted the AP1000 as the reference plant and hence the major milestones of the EPP program are related to the U.S. AP1000 program (Refer to this section in the AP1000 design description for additional information regarding AP1000 project status and schedule). This approach provides a cost effective way to achieve the objective of the EPP Agreement; i.e., a 1000 MW(e) passive PWR design based on passive technology that meets the EUR and is licensable in Europe. The EPP participants are contributing to development of AP1000 and its assessment according to EUR requirements.

4.12 SWR-1000 (FRAMATOME ANP, GERMANY)

4.12.1 Introduction

In 1992, German utilities awarded FRAMATOME (former Siemens) a contract to develop a new BWR nuclear power plant using passive safety systems, and together with the utilities FRAMATOME started development work on a new BWR with a net capacity of 750 MW(e). In the conceptual phase that lasted from February 1992 until September 1993, priority was given to developing passive safety systems to replace or supplement active systems. At the end of the conceptual phase, it was decided that the new requirements for this advanced BWR, especially economic aspects, justified a concept with a higher power output:

- Reactor thermal output 2778 MW
- Net electric output 977 MW

The four-year basic design phase for the resulting "SWR 1000" plant started in mid-1995. In parallel, an experimental testing program was conducted at FRAMATOME's own testing facilities and at other German and European research centers to provide verification of the mode of operation and effectiveness of the SWR 1000's passive safety systems.

Since 2000 an extended basic design phase has been underway. At the beginning of this phase it was decided to increase the net electric output to 1254 MW to serve the needs of the nuclear industry with a larger power range.

The main goal of this advanced BWR is to replace the active safety systems used in current designs with passive safety systems enabling:

- Reliable control of the various design basis accidents;
- Low probability of beyond-design-basis accidents (core damage frequency);
- Limitation of the consequences of a core melt accident to the plant itself;
- High plant availability;
- Economic competitiveness.

The adoption of passive safety systems requires a lot of engineering effort, planning and layout work to modify previous BWR system designs. The passive safety systems replacing and/or supplementing the redundant active safety systems must be capable of ensuring reliable plant operation and accident control. They mainly operate using basic laws of physics such as gravity, natural convection, temperature and pressure differentials.

Various features have been changed compared to existing BWR designs:

- Large water inventory in the reactor pressure vessel (RPV) above the core permits passive core cooling;
- Large water storage capacities inside and outside the reactor containment provide long grace periods and avoid the need for operator intervention, especially during and after accidents;
- For transients as well as for accident control, emergency condensers and containment cooling condensers passively remove decay heat from the core and containment, respectively;
- Activation of key safety functions such as reactor scram, containment isolation and automatic depressurization is backed up by passive systems (passive pressure pulse transmitters);
- Passive cooling of the RPV exterior in the event of a core melt accident ensures in-vessel melt retention;
- Despite the introduction of passive safety systems for accident control the operating experience gained from current BWR plants constitutes the basis for the new concept;
- Simplification of reactor auxiliary systems and systems used for normal power operation.

The new, innovative design features of the SWR 1000 mark the transition to the next generation of nuclear power plants.

4.12.2 Description of the nuclear systems

4.12.2.1 Reactor coolant system and its main characteristics

The reactor coolant system is located in the reactor building and is surrounded by a reinforced-concrete containment with steel liner.

Three main steam lines connecting the RPV to the high-pressure turbine section serve to transport the steam generated in the reactor to the turbine. Isolation valves of diverse design are provided in each main steam line inside and outside the containment. The inboard isolation valves are gate valves while the outboard isolation valves are angle-type globe valves.

The function of the feedwater system is to receive the water from the main condensate system and to route it to the reactor via two feedwater lines. Isolation valves are provided in each feedwater line inside and outside the containment. The inboard isolation valves are check valves while the outboard isolation valves are gate valves.

The outboard valves in both the main steam lines and the feedwater lines are located immediately adjacent to the containment.

Each main steam line inside the containment is allocated a specific number of safety-relief valves (SRVs) for overpressure protection of the RPV.

For pressure relief, opening of the SRVs is possible when a signal is received from the reactor protection system. If reactor pressure should, however, continue to rise, then all SRVs are mechanically opened by spring-loaded pilot valves. The SRVs close, however, with a corresponding hysteresis.

For automatic depressurization, the SRVs are opened either by solenoid pilot valves or by the passive pressure pulse transmitters (PPPTs) and diaphragm pilot valves, all SRVs being opened at the same time in this case. Half of the SRVs are designed such that they do not reclose after automatic depressurization. This ensures that the pressure in the reactor remains at a low level.

The steam blown down by the valves is routed downwards into the core flooding pools through discharge pipes. These pipes terminate below the surface of the pool water in specially designed T-shaped quenchers.

4.12.2.2 Reactor core and fuel design

The SWR 1000 core represents an "evolutionary" development based on previously common BWR core designs. While no changes have been made to the basic core structure, certain modifications have been introduced. These include a reduced active core height and an enlarged fuel assembly.

A consequence of reducing the active core height is that the core can be positioned lower down inside the RPV. This provides a larger water inventory inside the RPV above the core, a feature which facilitates accident control.

The above-mentioned modification of the fuel assemblies consisted of enlarging the existing ATRIUM^{TM}10 fuel assembly design (10x10-9Q) to a 12x12-16 rod array (ATRIUM^{TM}12). Fuel rod diameter and pitch, on the other hand, remained the same as in the ATRIUM^{TM}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 reduces the number of control rods, and hence the number of control rod drives as well. The average power density is around 51 kW/l.

4.12.2.3 Fuel handling and storage systems

There is no significant difference from a functional point of view between the equipment and structures used for refueling, storage of new and spent fuel assemblies and handling of reactor components in the SWR 1000 and those found in traditional BWR nuclear power plants.

New fuel store

New (i.e. unirradiated) fuel assemblies are stored in a new fuel store specially provided for this purpose adjacent to the spent fuel pool. The new fuel assemblies are placed in dry storage racks which can accommodate approximately 144 fuel assemblies.

Spent fuel pool

Spent fuel assemblies are stored in the water-filled fuel pool located inside the reactor building on an extension of the axis of the shielding/storage pool and reactor well. The pool water provides for residual heat removal and shielding. The fuel pool has sufficient storage capacity for approximately 1650 fuel assemblies and approximately 50 control rods.

4.12.2.4 Main components

Reactor pressure vessel

The RPV surrounds the reactor core and the RPV internals. As part of the reactor coolant pressure boundary (RCPB), the RPV represents an important confinement barrier for preventing the release of radioactive materials. Figure 4.12-1 shows the RPV together with its internals and insulation as well as the inner concrete cylinder of the drywell surrounding it.

The RPV consists of a cylindrical section with a spherically dished bottom head, and a hemispherical top head that is flanged to the cylindrical section. The vessel flange protrudes mainly inwards. The vessel top head is bolted to the lower section of the vessel by means of studs.

Welded to the outside of the RPV is a support skirt which accommodates all static and dynamic loads, transmitting them to the inner concrete cylinder of the drywell and thus to the concrete foundation. The entire inside surface of the RPV shell is provided with weld-overlay cladding made of stabilized austenitic stainless steel. The pressure vessel is manufactured from e.g. SA 508 Gr. 3 Cl. 1. The alloy composition is specified within a very narrow range to optimize ductility, stresses and resistance to neutron embrittlement.

The RPV has numerous nozzles for connecting the piping of the main steam, feedwater, emergency condenser, shutdown cooling and vessel head spray systems, as well as for accommodating the internal reactor water recirculation pumps, the control rod drives, the core flux monitoring assemblies and the reactor water level, pressure and temperature measuring instrumentation.

The four emergency condensers as well as the four standpipes (which connect the passive pressure pulse transmitters and the condensation pots of the RPV level measuring equipment to the reactor vessel) are regarded as being external extensions to the vessel since they are connected to it via nonisolatable lines. These components are subject to the same requirements and conditions as those applying to the RPV.

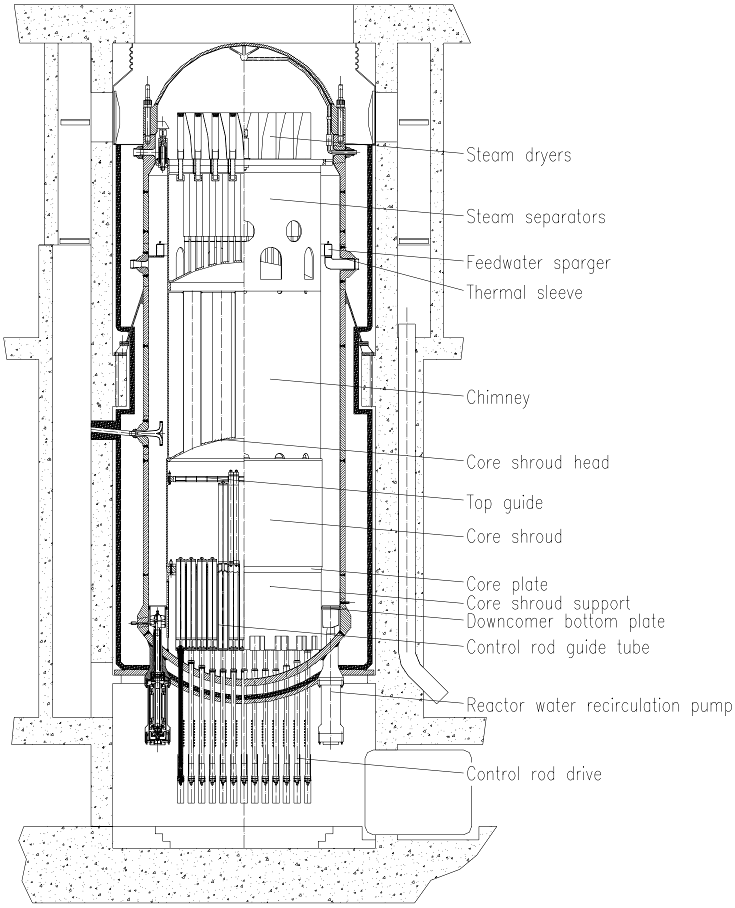


FIG. 4.12-1. SWR 1000 reactor pressure vessel and internals.

Reactor internals

The main reactor internals are illustrated in Figure 4.12-1. The RPV internals primarily serve to:

- Separate the regions containing coolant from those containing steam;
- Enhance internal circulation of the reactor coolant;
- Provide the necessary flow paths for the circulating coolant;
- Ensure uniform distribution of the feedwater in the downcomer annulus;
- Guide the control rods and the core instrumentation assemblies;
- Direct the flow of the steam-water mixture produced in the reactor core;
- Locate the fuel assemblies in the core;
- Perform phase separation of the steam-water mixture;
- Dry the steam inside the RPV.

In designing the RPV internals, preference was given to using components and component designs that had already proven their reliability during operation at plants belonging to Siemens' 1969 and 1972 Product Lines. Designs deviating from these were only selected if they resulted in shorter refueling outages or if they were unavoidable due to the large steam-water plenum above the core (i.e. the space between the low-elevation core and the relatively high-elevation steam separators).

All RPV internals are designed for removal and replacement as necessary.

Reactor water recirculation pumps

The eight reactor water recirculation pumps ensure sufficient cooling of the reactor core. Reactor power output can be controlled by varying the speed of the pumps. If one of the pumps should fail, the flow of coolant through the core can be kept constant by increasing the speed of the remaining pumps.

The pumps that have been selected for the SWR 1000 are so-called wet-motor pumps in which the electric pump motor is situated inside the reactor coolant pressure boundary. This eliminates the need for mechanical seals as well as for an oil-lubricated, combined thrust and journal bearing with its dedicated oil supply system. Moreover, there is no longer any axial thrust induced by the reactor coolant pressure that has to be accommodated.

4.12.2.5 Reactor auxiliary systems

The main auxiliary systems of the SWR 1000 are

- the residual heat removal (RHR) system;
- the reactor water cleanup (RWCU) system and fuel pool cleanup system;
- the fuel pool cooling system.

Apart from these systems many other auxiliary systems such as waste processing systems, a chilled water system and HVAC systems, etc. exist for normal operation of the plant.

The SWR 1000 design concept includes two active systems for low-pressure core injection/flooding and RHR. As in earlier plant designs these systems perform the following tasks:

- Cooling of the reactor core during and after normal plant shutdown;
- Water transfer operations before and after refueling;
- Operational heat removal from the core flooding pools and the pressure suppression pool;
- 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 coolant injection into the RPV and simultaneous heat removal in the event of loss-of-coolant accidents.

The systems are actuated by safety instrumentation and control (I&C) equipment and their electrical loads are connected to an emergency power supply system.

In the SWR 1000, unlike in previous BWR plants, a low-pressure concept (standard practice in PWR plants) is applied to the reactor water cleanup system outside the containment. A regenerative heat exchanger and a pressure-reducing station are arranged inside the containment, while another cooler, the cleanup pumps, the reactor water filter-demineralizers and the filter precoating station are all located outside the containment. The purified water is conveyed back into the containment by one of two return pumps and fed to the RPV via the regenerative heat exchanger. The two return pumps also supply cooling water to the control rod drives and seal water to the reactor water recirculation pumps.

The advantage of this concept lies in the fact that the filter-demineralizers, 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 filter-demineralizers, now located in the reactor building, can also be used for cleanup of the fuel pool water and the water in the shielding/storage pool.

The fuel pool must be cooled, independently of reactor operation, for as long as the removal of decay heat generated by the spent fuel assemblies in the fuel pool is required. In existing BWR plants two fuel pool cooling trains are installed for purposes of decay heat removal.

In contrast to this, the fuel pool of the SWR 1000 is equipped with two redundant cooler units, each consisting of four heat exchangers operating in parallel, which are installed directly in the fuel pool. The fuel pool water is cooled by means of natural convection. Redundancy is ensured by connecting the cooler units to the redundant closed cooling water systems which are backed up by an emergency power supply. The tubular heat exchangers are suspended from the fuel pool wall such that controlled water flow conditions are obtained.

4.12.2.6 Operating characteristics

The power plant is controlled using two separate control loops: reactor power is matched to present power requirements by the reactor power controller; the turbine is controlled by a constant main steam pressure being maintained upstream of the turbine by the reactor pressure controller.

The main mechanism employed for reactor power control is that of recirculation flow control which changes the flow of coolant through the core. The control rod follow-up control system keeps the reactor water recirculation pumps in a favorable speed range. The pressure in the reactor is maintained at a constant level using the turbine control valves which adjust the main steam flow accepted from the reactor.

Automatic control of reactor power is possible over the range between 20 and 100% of rated power. Reactor power can be adjusted by about 30% by changing the core coolant flow rate without repositioning control rods; for example, from 100% to around 70% of rated power. If required, reactor power can be further reduced in increments down to approximately 40% of rated power by shutting down between one and a maximum of eight (all) reactor water recirculation pumps. The speed of the pumps being shut down is automatically reduced to zero by their mechanical brakes.

For the control rod patterns employed for operation at reduced power, the control range that can be achieved by altering coolant flow rate extends between around 0.7 and 1.0 times the selected power level. In this way automatic control of reactor power is possible down to a level of about 40% of thermal output. The core coolant flow rate is changed by adjusting the speed of the reactor water recirculation pumps.

The speed of the pumps can be manually controlled from the control room. After this form of control has been selected, the speed of all pumps can be changed by adjusting the power setpoint.

4.12.3 Description of turbine generator plant systems

4.12.3.1 Turbine generator plant

The turbine is connected to the RPV by three main steam lines. The bypass station that is installed in parallel to the turbine enables 60% of the steam produced inside the reactor to be dumped directly to the condenser. The turbine generator set consists of a single-shaft saturated-steam turbine coupled directly to a three-phase AC synchronous generator. The 1500-rpm turbine comprises one high-pressure (HP) and three low-pressure (LP) turbine sections. Both the generator stator and rotor are hydrogen-cooled. The main steam from the reactor is admitted to the HP turbine section via combined stop and control valves.

After undergoing partial expansion in the HP turbine section, the steam passes through a moisture separator with reheater to the three double-flow LP sections of the turbine. The steam expands in the LP sections through several stages of blading down to condenser pressure.

4.12.3.2 Condensate and feedwater systems

The condensate collected in the hotwells of the three condensers is discharged into a common header supplying the main condensate pumps. The condensate passes through a demineralizing system before being heated to final feedwater temperature in feedwater heaters. The feedwater is fed to the reactor via two feedwater lines by variable-speed feedwater pumps.

4.12.3.3 Auxiliary systems

The condensate demineralizing system is situated between the two main condensate pumps and the LP feedwater heaters and is installed in the reactor auxiliary building.

The main condensate is passed through the condensate demineralizing system after leaving the turbine condensers. In this system even very small quantities of salts, which may have entered the water via tiny leaks in the turbine condenser, are removed in addition to corrosion products from the steam, condensate and feedwater cycle. The condensate demineralizing system comprises five filters arranged in a parallel configuration. Each filter tank is housed in a compartment providing shielding.

4.12.4 Instrumentation and control systems

4.12.4.1 Design concept, including control room

The digital I&C concept planned for the SWR 1000 is made up of the following subsystems:

- Operational I&C system,
- Safety I&C system,
- Screen-based man-machine interface.

Operational I&C encompasses all systems required for process control during normal operation (i.e. during power operation and in the shutdown condition), such as:

- The process automation system, including sensors, automatic controls and component protection functions,
- The process information and control system, including the man-machine interface and the remote shutdown station,
- The plant bus and terminal bus system,
- The diagnostic and engineering system.

Main control room

The control room provides for central control of power plant operation. It houses the operator control and information display equipment for managing and monitoring of the plant.

The design of the control room and its technical equipment is based on the assumption that a nuclear power plant is equipped with extensive, automated I&C equipment.

The technology implemented in the control rooms of today's nuclear power plants as well as in control rooms of the future is and will be characterized to a large extent by screen-based displays and screen-based control equipment. The design concept of the SWR 1000's control room is therefore based on a powerful process information system for operating data analysis and storage, trend analyses, preventive maintenance, process optimization, support of the operating personnel in fault analysis and other tasks.

Remote shutdown station

The remote shutdown station is used to control the power plant in the event of loss of the main control room. The control room and the remote shutdown station are designed such that a simultaneous loss of both areas is not possible.

Process control can be switched over from the control room to the remote shutdown station and back again, as required. However, individual power plant processes can only be controlled from one of these locations at a time.

Technical support center

The technical support center serves as an area in which information is supplied in the event of an accident to a team of technical advisors. Intervention in plant operation is not possible from the technical support center.

The center is physically separated from the control room. It is linked to the plant I&C systems via computer monitors and is designed for occupancy by 10 persons.

Local panels

Local control stations are provided for tasks that are not directly linked to power generation but which require close proximity to plant processes. These control stations are only used when local control is necessary. Examples are the emergency diesels and the sampling system.

4.12.4.2 Reactor protection and other safety systems

The task of the safety I&C systems is to process and monitor key process variables important to reactor safety and environmental protection in order to detect accident conditions and, as a supplement to passive switching operations, to automatically initiate safety functions for maintaining reactor conditions within safe limits. Safety I&C initiates no actions during normal plant operation but takes priority over all operational I&C system functions on demand.

The computer-based process information system is a global information source. Intelligent information processing and reduction enable it to display process conditions and process sequences with a high information content for safety-related and operational tasks.

The safety-related alarm and information system first and foremost informs the operating personnel about the status and condition of plant safety equipment.

The various I&C subsystems are connected to each other via a common, redundant plant bus.

The safety systems of the SWR 1000 consist of both active and passive safety systems. As the passive safety systems alone are capable of controlling all postulated accidents during power operation, the safety I&C equipment can be limited to a redundancy of two 100%-capacity trains. This two-train configuration is maintained throughout all I&C, process and power supply systems. The process variables to be monitored are recorded in each train by means of two measuring transducers. Further processing of measuring signals, limit value generation, formation and selection of actuation signals, and logic gating to generate trip signals are all performed by means of a digital system. The trip signals directly actuate a process component or component group.

The provision of 12 passive pressure pulse transmitters (PPPTs) represents a new design feature of the SWR 1000. The function of the PPPTs is to automatically initiate switching operations without any external source of power in the event that the water level in the reactor vessel drops below or exceeds predefined limits. In the case of accident sequences which do not involve an immediate drop in reactor water level but a large rise in pressure, switching operations can be initiated automatically by a PPPT without any external power in response to violation of the pressure limit.

4.12.5 Electrical systems

4.12.5.1 Normal power supply systems

The auxiliary power supply system and the offsite power system connection correspond to today's well-proven technology.

The generator feeds power into the main offsite power system via the generator transformer and the main offsite power system connection. Auxiliary power is tapped off between the generator breaker and the offsite power system breaker and fed to the auxiliary power supply system via two three-winding auxiliary power transformers with on-load tap changers.

The auxiliary power busbars are additionally connected to the offsite power system via a standby auxiliary power transformer and a standby offsite power system connection. In the event of unit failure, or loss of the main offsite power connection, the supply of auxiliary power can nevertheless be ensured by switching over to the standby offsite power system connection.

In view of the different power levels of the various loads connected to the plant's auxiliary power supply system, three voltage levels are provided.

For reasons of maintaining voltage stability, only electrical loads with ratings of < 1 MW are connected to the 10-kV busbars to which the emergency power supply trains are connected. Electrical loads having higher ratings are connected to the second 10-kV busbar of each train.

One 690-V emergency power busbar per train is supplied from the 690-V busbar of the auxiliary power supply system.

4.12.5.2 Safety-related systems

All electrical loads which must remain in operation or must be started up in the event of loss of normal power are connected to the emergency power supply system.

In the event of loss of normal power, an emergency diesel generator takes over independent power supply to all connected electrical loads. Electrical loads for which a period without power is allowable during runup of the emergency diesel generator are connected to the three-phase AC distribution boards of the emergency power supply system.

Electrical loads which, upon loss of normal power, must remain in operation without interruption or must be immediately started up are connected to the uninterruptible power supply (UPS). These loads are supplied with power either directly from the 220-V DC system or indirectly via the downcircuit inverters and the connected 400-V distribution board, or directly from distributed UPS equipment.

The two 220-V DC systems in each power supply system train are each supplied with power via a separate rectifier from the 690-V emergency power distribution board of their respective train. Each 220-V DC system is equipped with a battery and a battery charger.

The chargers have sufficient capacity to supply the trickle charge needed to keep the battery fully charged, the power required by the loads that are connected to only one power supply train, and also the power required by the loads powered from diode-decoupled power supplies in both their own and the other train.

The batteries are used to supply power during emergency diesel startup following loss of normal power and to also supply the required power in the event of failure of the battery chargers. They are operated in the trickle-charging mode. The batteries are designed such that they can continue to supply the DC loads without interruption from the moment the auxiliary power supply is lost until the load is picked up by the emergency diesel generators and the chargers are reconnected. If the chargers should not be available, the batteries can continue to supply the power requirements of the connected loads for at least two hours.

The 400/230-V inverters are used for providing an uninterruptible power supply to three-phase and single-phase AC loads. They are each supplied from the 220-V DC busbar of their respective system.

Upon failure of an inverter, its downcircuit 400-V distribution board continues to be supplied with power without any interruption via an electronic transfer switch from the 400-V emergency power distribution board of the same train.

The I&C systems are supplied with power at a constant voltage of ± 24 V by the 220-V DC systems via DC/DC converters. The power is supplied from two diode-decoupled power supplies: from a 220-V DC system belonging to the equipment's own train and from a 220-V DC system in the second power supply system train.

Power is supplied to the monitoring systems and computers at the master control console in the main control room and in the remote shutdown station via inverters or UPS equipment.

4.12.6 Safety concept

4.12.6.1 Safety requirements and design philosophy

The safety concept of the SWR 1000 is based on two fundamental principles:

1) Prevention of accidents and environmental impacts
2) Accident control (damage mitigation)

First Safety Principle: "Prevention of Accidents and Environmental Impacts"

This first and most important principle is put into practice by imposing stringent requirements on the design and quality of the plant as well as on the qualifications of personnel, i.e. their competence and reliability.

For this purpose, safety-promoting design, manufacturing and operating principles are pursued on the **first level** of safety measures.

According to general experience with technical systems, malfunctions of components or systems leading to offnormal operating conditions cannot be entirely ruled out during the service life of a plant, even if the above principles have been observed. In order to control these offnormal operating conditions, systems are designed and measures are taken to control and monitor operation such that the occurrence of accidents as a consequence of offnormal operating conditions is prevented with an adequate degree of reliability (**second level**).

Second Safety Principle: "Accident Control (Damage Mitigation)"

Despite the precautions taken in compliance with the first safety principle on the two levels described above, it is nevertheless assumed that improbable but hypothetically conceivable accidents may occur during the service life of the nuclear power plant, i.e. accidents which the plant must be designed to control. These accidents – called design basis accidents – include, for example, the following:

- Accidents caused by plant-internal events: main steam line break, feedwater line break, control equipment malfunctions or accidents not involving loss of coolant;
- Accidents due to natural or external man-made hazards: earthquakes or high water levels.

In order to fulfill the second principle, equipment for accident control is provided on a **third level** of safety measures. In the case of the SWR 1000, new approaches have been pursued which lead to a significantly higher level of safety.

The safety equipment is designed in such a way that it protects the plant personnel and the population in the vicinity of the plant against the consequences of accidents. For this, the following design principles are applied:

- Redundancy, diversity and independence of subsystems (trains);
- Physical separation of redundant subsystems (trains);
- Safety-oriented system behavior in the case of subsystem or component malfunctions;
- Passive safety functions given preference over active functions.

Equipment for accident control consists of passive and active safety components.

Passive components, which do not require I&C signals or external power to perform their protective function, take effect solely by virtue of their presence (such as the numerous protective barriers made of concrete or steel) or as a result of basic laws of physics (such as gravity and natural convection). Examples of such equipment are the emergency condenser and the passive pressure pulse transmitter.

Active safety equipment, such as the RHR pumps and the control rods, are controlled and, if necessary, put into operation by the reactor protection system.

In addition to the measures for controlling design basis accidents, features are also provided on a **fourth level** of safety measures to mitigate the consequences of severe accidents, such as:

- Aircraft crash,
- Explosion pressure waves,
- Combustible and toxic gases,
- Core melt.

4.12.6.2 Safety systems and features (active, passive and inherent)

Passive equipment for accident control

The fundamentally new concept for accident control incorporated into the SWR 1000 includes equipment which, in the event of failure of the active safety equipment, will bring the plant to a safe condition without the need for any I&C signals or external power. This passive safety equipment (Figure 4.12-2) includes the following:

- *Emergency condensers*
  The function of the emergency condenser (EC) system is to remove, in the event of an accident, the decay heat still being generated in the reactor as well as any sensible heat stored in the RPV to the core flooding pools, without any coolant inventory being lost from the RPV. The system thus replaces the high-pressure coolant injection systems used in existing BWR plants. The EC system also provides a means for reactor pressure relief that is diverse with respect to the safety-relief valves.

- *Containment cooling condensers*
  The task of the four containment cooling condensers (CCCs) is to remove — by entirely passive means — decay heat from the containment following accidents leading to the release of steam inside the drywell, and in this way to limit buildup of containment pressure. They provide redundancy and diversity with respect to the RHR system.

- *Core flooding system*
  The core flooding system is a passive low-pressure flooding system for controlling the effects of loss-of-coolant accidents (LOCAs). It is installed at an elevation which ensures that, following automatic depressurization of the reactor, it can passively flood the reactor core by means of gravity flow. The system provides redundancy and diversity with respect to the core flooding function of the RHR system.

- *Drywell flooding system*
  A postulated severe accident involving core melt is controlled such that the molten core is retained inside the RPV. For this purpose the section of the drywell surrounding the RPV is flooded with water in order to cool the RPV exterior and thus remove heat from the reactor.

- *Passive pressure pulse transmitters*
  The PPPT is a completely passive switching device which is used to directly initiate the following safety functions (as a minimum), without the need for I&C equipment: reactor scram, containment isolation at the main steam line penetrations, and automatic depressurization of the RPV. The PPPT comes into action as a result of a drop or increase in reactor water level as well as an increase in reactor pressure. For activating the various safety functions, PPPTs of redundant design are installed at two elevations. The upper PPPTs, situated at an elevation beneath that of the normal water level of the RPV, are responsible for initiating reactor scram. The lower PPPTs, arranged at a lower elevation, activate automatic depressurization of the reactor as well as closure of the main steam containment isolation valves. Further PPPTs installed at appropriate locations respond to a rise in reactor water level above the main steam nozzles and likewise activate containment isolation at the main steam line penetrations.

Active safety systems

In order to control the effects of design basis accidents, each nuclear power plant is equipped on the third safety level with a special safety system (Figure 4.12-2) consisting of the reactor protection system and the active safety equipment actuated by it. The reactor protection system is a programmable digital I&C system which continuously monitors all important plant operating parameters, initiates safety-oriented actions if specified limits are approached and, in this way, takes control over operational disturbances, thus preventing them from developing into accidents. The postulated design basis accidents can thus be controlled to such an extent through activation of the safety equipment that consequences are restricted to the plant itself.

The response of the reactor protection system is not event-oriented but safety-oriented, something which ensures that no potential causes of failures or malfunctions can be overlooked when designing the system. The active safety equipment mainly comprises process systems.

As far as the overall safety concept is concerned, it is vitally important that all types of accidents which involve the risk of an increased release of activity to the environment be determined.

By far the largest proportion of radioactivity present in the nuclear power plant is located in the reactor core, i.e. contained in the crystal lattice of the fuel and in the fuel cladding tubes. Therefore large releases are only conceivable if these two inner activity barriers should become damaged. The following theoretically conceivable types of accidents involving the risk of an increased release of activity are therefore possible in the event of damage to these two barriers:

- Unallowable rise in reactor power,
- Impaired heat removal from the reactor core,
- Loss of cooling as a result of a loss of coolant.

Among the various active safety systems, a central role is played by the reactor protection system which continuously monitors all important plant operating parameters and, if specified limits are approached, initiates safety actions by actuating other safety equipment as and when required.

The hydraulic scram system employs neutron-absorbing control rods which are kept in the withdrawn position, i.e. at the bottom of the core, during reactor power operation. If a scram is triggered, valves are opened in lines leading to the scram tanks and the energy stored in these accumulators rapidly inserts the control rods into the core from below, thus terminating the chain reaction.

A second, diverse shutdown system is also provided with which the reactor can be shut down by injecting a neutron-absorbing boron solution into the reactor coolant.

If a release of radioactivity into the containment is to be expected during an accident, the containment isolation system allows the containment to be isolated from the plant environs. All piping which penetrates the containment wall and belongs to systems not required for accident control can be isolated by containment isolation valves.

The RHR system takes over cooling of the reactor core and/or containment heat removal in the event of an accident.

Finally, mention should be made of the emergency power supply system which supplies power to active safety-related systems if the main generator cannot provide auxiliary power in the event of an accident and if supply from the offsite power system is not available.

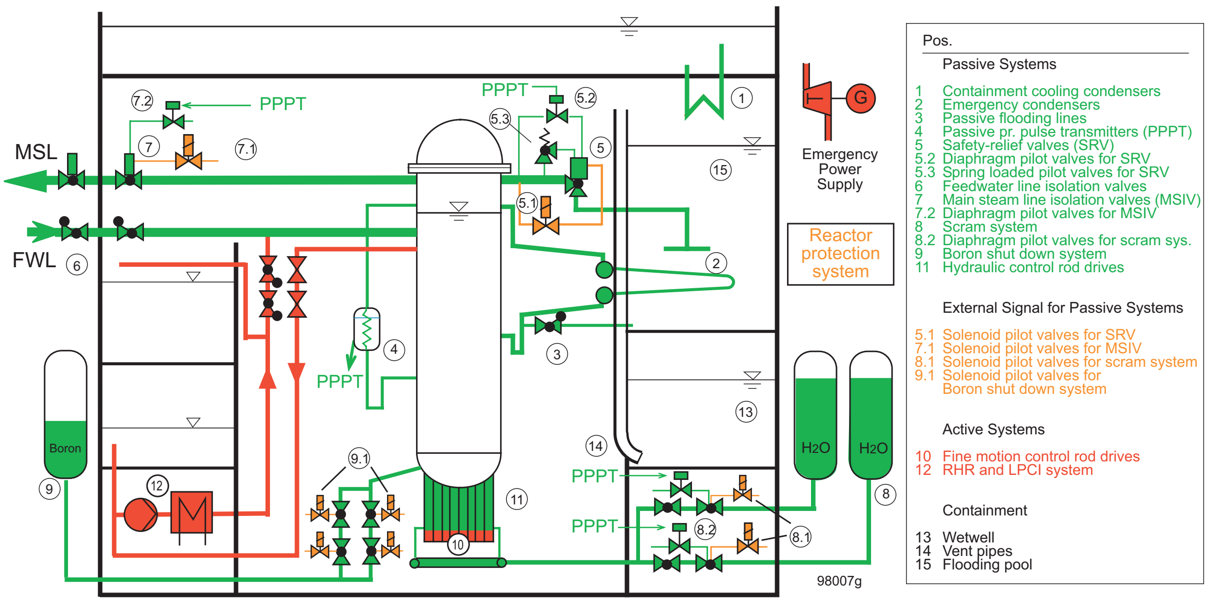


FIG. 4.12-2. SWR 1000 – Active and passive systems.

4.12.6.3 Severe accidents (beyond design basis accidents)

Severe accidents are those accidents which could lead to core damage states (core melt). For new nuclear power plants it must be ensured that the consequences of such accidents would be limited to the plant itself and that there would be no need for large-scale actions to protect against the harmful effects of ionizing radiation beyond the plant perimeter. Therefore severe accidents are likewise taken into account in plant design despite their extremely low frequency of occurrence. In view of the above requirements, the aim of severe accident analysis is to verify that, even if the core should become damaged, no unallowable releases of radioactivity to the plant environs will result.

Loss of all active and passive means of supplying coolant to the RPV as well as of all emergency condensers is assumed for the most severe postulated accident: core melt. To control this accident scenario additional safety systems are planned for the SWR 1000 and the plant is designed to withstand the consequences of the accident.

Core melt at high pressure is ruled out by the design of the depressurization system. The melt is retained inside the reactor vessel under low-pressure conditions due to cooling of the RPV exterior. A flooding system is installed for this purpose which feeds water from the core flooding pools into the section of the drywell surrounding the RPV. The drywell flooding line is normally isolated, the valves in this line only being opened when flooding is required. The steam arising during cooling of the RPV exterior is condensed at the containment cooling condensers which transfer the heat from the containment to the water in the shielding/storage pool. The capacity of the shielding/storage pool is such that makeup water does not need to be supplied to the pool until several days after onset of the accident, thus enabling 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 some of this hydrogen is also carried over into the pressure suppression chamber, depending on the given pressure conditions. Any further pressure buildup due to chemical reactions by the hydrogen is not possible since the containment is inerted with nitrogen.

4.12.7 Plant layout

4.12.7.1 Buildings and structures, including plot plan

The systems and components are arranged inside the various plant buildings and structures (Figure 4.12-3) in clearly delineated structural complexes, thereby enabling the buildings to be constructed in parallel. The reactor building and the turbine building form the central complex of the plant. Other buildings are the switchgear building, the reactor auxiliary building, the reactor supporting systems building, the emergency control room building, the circulating water pump building, service water pump building and emergency diesel generator building, and the complex with the second service water pump building and emergency diesel generator building.

In the arrangement of the buildings, a distinction is made between site-specific buildings and structures such as those for the circulating/cooling water supply systems, and non-site-specific buildings such as the following:

- Reactor building, including containment,
- Turbine building,
- Reactor auxiliary building,
- Reactor supporting systems building,
- Vent stack,
- Emergency diesel generator buildings,
- Service water pump buildings
- Emergency control room building.

Piping and cables between the buildings are either buried underground or routed in ductwork.

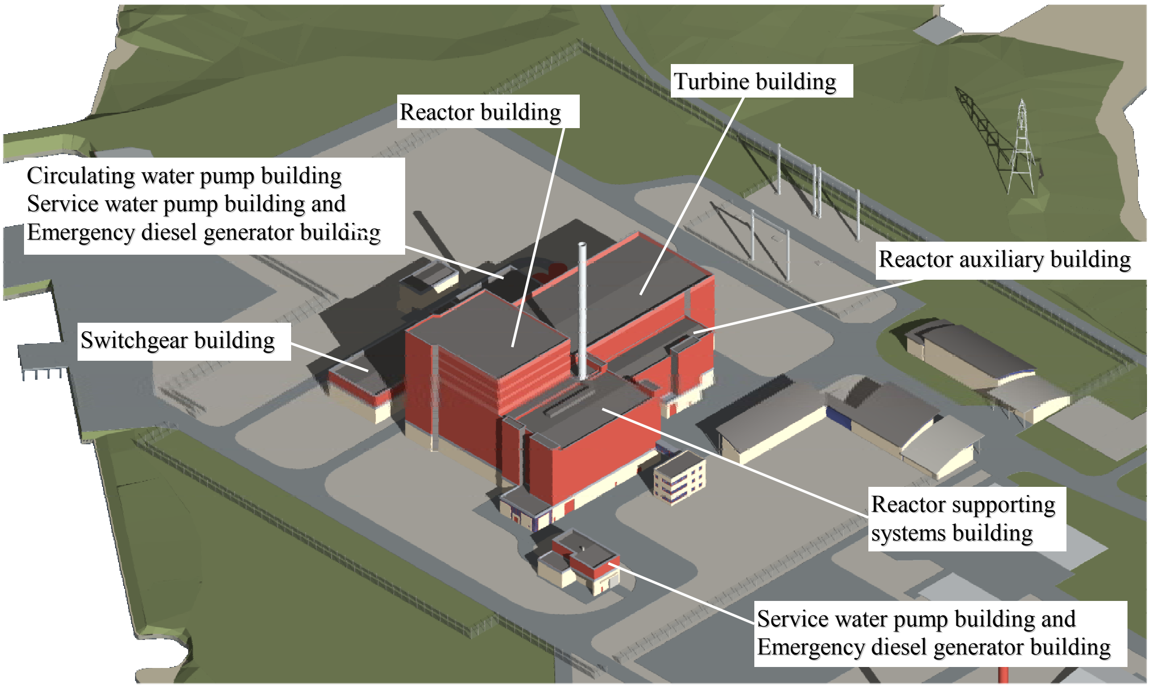


FIG. 4.12-3. SWR 1000 – Typical site layout

4.12.7.2 Reactor building

The reactor building houses the containment and the safety-related mechanical components, the safety-related electrical and I&C equipment, and the required power supply (batteries) and protection systems. It provides protection against natural and external man-made hazards (including aircraft crash) and serves as a confinement barrier for radioactivity in the event of accidents.

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
- Containment.

4.12.7.3 Containment

The primary function of the containment is to protect the environment against any release of radioactive materials and against direct radiation under all possible accident conditions.

A cylindrical containment made of steel-reinforced concrete equipped with an inner steel liner and a pressure suppression system has been selected (Figure 4.12.5). In keeping with its pressure-suppression-type design the containment is divided into a drywell and a pressure suppression chamber (wetwell).

The containment is also designed to accommodate the pressure buildup due to the hydrogen produced by a 100% zirconium-water reaction, i.e. involving the entire zirconium inventory present in the core, in the event of a core melt accident.

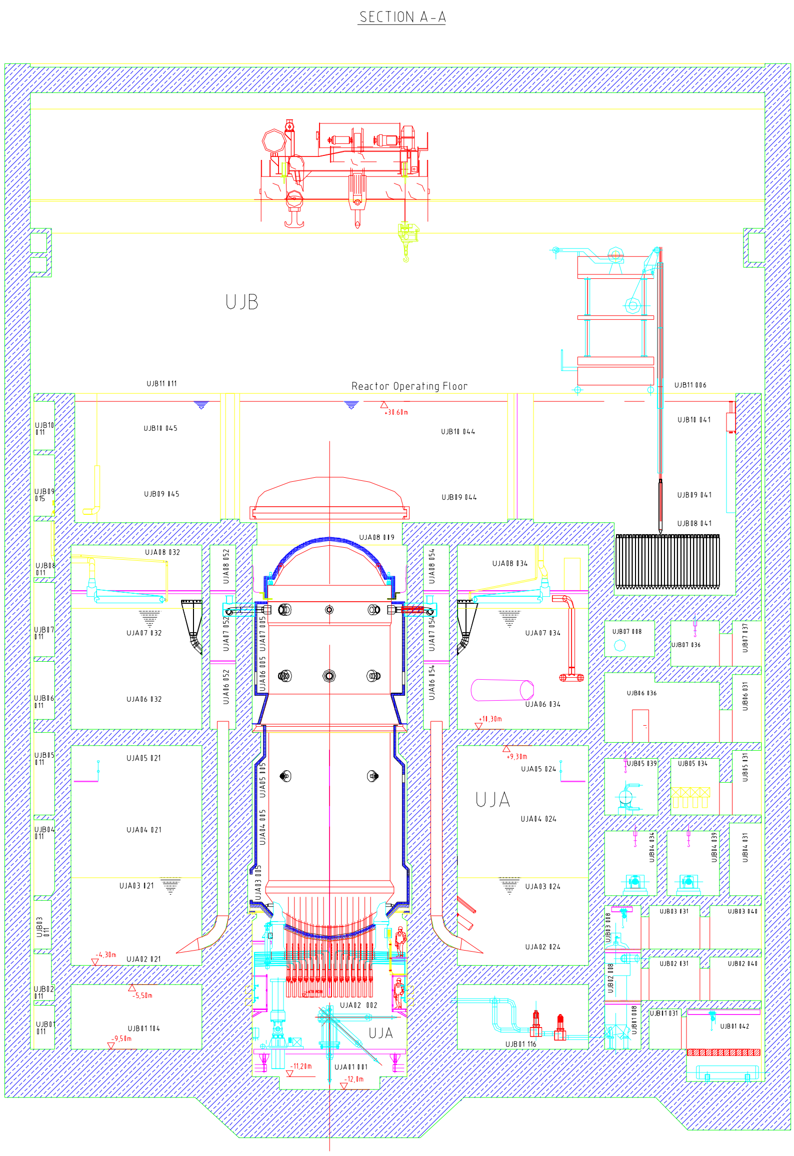


FIG. 4.12-4. SWR 1000 - Reactor building; longitudinal section A-A.

Drywell

In addition to the RPV and the three main steam lines and two feedwater lines, the following components are also located in the drywell: four large hydraulically interconnected 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 activating safety functions without any need for I&C signals. In addition, the drywell is equipped with two full-capacity recirculation air cooling systems. The HP section of the reactor water cleanup system (HP cooler and pressure-reducing station) as well as the lines of the RHR system are also situated inside the drywell. Thanks to the shorter control rod drives, the RPV can be positioned lower down inside the containment.

The entire containment is inerted with nitrogen during normal operation. This not only reliably prevents any hydrogen-oxygen reactions in the event of a core melt accident, but also provides fire protection under normal plant operating conditions.

The RHR pumps and heat exchangers are installed in separate compartments located underneath the pressure suppression chamber which are not inerted and are accessible from the outside at all times for maintenance and repairs.

One of the key differences between the containment of a standard BWR plant and that of the SWR 1000 lies in the latter's ability to store decay heat inside the containment (thanks to the larger water inventory) over a longer period of time. As a result it is not necessary for operating personnel to actively intervene until several days after onset of an accident.

Pressure suppression chamber

The pressure suppression chamber performs the following tasks:

- Serves as a heat sink in the event of an accident,
- Provides water for active RPV coolant makeup via the RHR system.

As part of the pressure suppression system, the pressure suppression chamber is located between the outer and inner containment cylinders beneath the core flooding pools (Figure 4.12-5) and is one-third filled with water. The pressure suppression chamber is connected to the drywell via vent pipes embedded in the concrete of the inner cylinder. In addition, the pressure suppression chamber and core flooding pools are connected to each other via submerged water overflow and hydrogen overflow pipes. Connections between the drywell and the air space of the pressure suppression chamber known from existing BWR plants, such as pressure-equalizing valves, have been eliminated.

Core flooding pools

The interconnected core flooding pools act as a heat sink for the emergency condensers and the safety-relief valve system. In addition, owing to the pools' 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. For this function, spring-assisted check valves open the core flooding lines automatically. Passive core flooding serves as a diverse means of providing RPV coolant makeup which supplements the active core cooling systems.

In the event of a core melt accident, flooding pool water is used to cool the RPV from the outside.

The core flooding pools are located above the pressure suppression chamber and are approximately two-thirds filled with water. They are physically separated from each other by four equipment compartments containing mechanical components, piping and ventilation equipment. Each pool houses an emergency condenser, a containment cooling condenser (above the water surface), a core flooding line connection, and the SRV discharge pipes with steam quenchers. In addition, a drywell flooding line for cooling of the RPV exterior leads down to the bottom of the drywell.

4.12.7.4 Turbine building

The turbine building mainly contains the systems and components of the steam, condensate and feedwater cycle, such as the condensate and feedwater pumps and feedwater heaters as well as the turbine and generator.

The turbine building forms part of the controlled access area of the plant.

4.12.7.5 Other buildings

Reactor auxiliary building

The reactor auxiliary building contains systems and components used for processing and storage of liquid radioactive waste, including the evaporator system.

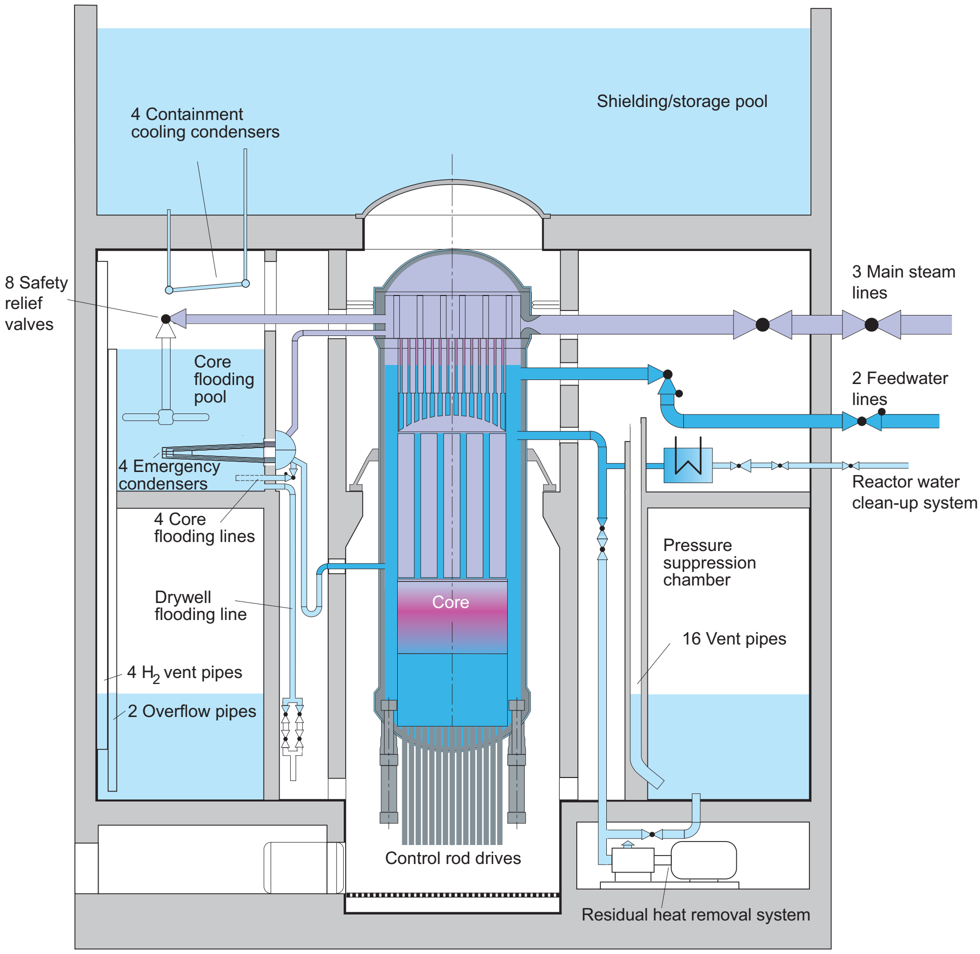


FIG. 4.12-5. SWR 1000 - Containment

The liquid waste processing and storage system is arranged such as to ensure short piping connections to the system and equipment areas in the turbine building and reactor building.

Reactor supporting systems building

The reactor supporting systems building contains the workshops and parts of the waste processing and storage system, as well as the central entrance to the controlled access area. This building houses components belonging to the following systems:

- Main control room;
- Intake and exhaust air system;
- Sanitary facilities, in particular the changing rooms and washroom facilities required at the entrance to the controlled access area;
- Laboratory;
- Hot workshop and decontamination facilities;
- Reserved space for mobile concentrates treatment unit with drum store for low-level waste;
- Non-safety-related switchgear.

4.12.8 Technical data

General plant data

Power plant output, gross	1290	MW(e)
Power plant output, net	1250	MW(e)
Reactor thermal output	3370	MWt
Power plant efficiency, net	37	%
Cooling water temperature	8	°C

Nuclear steam supply system

Number of feedwater lines	2	
Number of main steam lines	3	
Steam flow rate at nominal conditions	1840	kg/s
Feedwater flow rate at nominal conditions	1838	kg/s

Reactor coolant system

Coolant flow rate	13,200	kg/s
Reactor operating pressure	7.5	MPa
Steam temperature/pressure	290/7.5	°C /MPa
Feedwater temperature	220	°C
Core coolant inlet temperature	282	°C
Core coolant outlet temperature	290	°C
Mean temperature rise across core	8	K

Reactor core

Active core height	3.0	m
Equivalent core diameter	5553	m
Heat transfer surface in the core	8050	m^2
Average linear heat rate	12.7	kW/m
Fuel weight {UO_2}	137	t U
Average fuel power density	24.7	kW/kg U
Average core power density	51	kW/l
Thermal heat flux, F_q	40.2	
Enthalpy rise, F_H	-	
Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	3763	mm
Rod array	12x12-16Q	
Number of fuel assemblies	664	
Number of fuel rods/assembly	128	
Number of spacers	7	
Enrichment of first core, average	2.47	wt%
Enrichment of reload fuel at equilibrium core	3.54	wt%
Operating cycle length (fuel cycle length)	12	months
Average discharge burnup of fuel	45,000	MWd/t
Cladding tube material	Annealed, recrystallized Zry-2	
Cladding tube wall thickness	0.605	mm
Outer diameter of fuel rods	10.05	mm
Fuel channel material	Zry-2	
Overall weight of assembly	325	kg
Uranium weight/assembly	205	kg
Active length of fuel rods	3000	mm
Burnable absorber, strategy/material	Gd_2O_3 in fuel	
Number of control rods	157	
Absorber material	B_4C	
Drive mechanism	Electromechanical fine-motion	
Positioning rate	25	mm/s
Soluble neutron absorber	-	

Reactor pressure vessel

Inner diameter of cylindrical shell		7120	mm
Wall thickness of cylindrical shell		183+8	mm
Total height, inside		23,450	mm
Base material:	RPV lower section	SA 508 Gr. 3 Cl. 1	
	RPV top head	SA 508 Gr. 3 Cl. 1	
	cladding	Austenitic steel	
Design pressure/temperature		8.8/300	MPa/°C
Shipping weight	RPV lower section	850	t
	RPV top head	130	t

Reactor water recirculation pump

Type	Vertical centrifugal	
Number	8	
Design pressure/temperature	9.5/300	MPa/°C
Design mass flow rate (at operating conditions)	1683	kg/s
Pump head	37	kPa
Rated power of pump motor (nominal flow rate)	975	kW
Pump casing material	as for RPV	
Pump speed (at rated conditions)	1661	rpm

Containment

Type	Pressure-suppression	
Overall form	Cylindrical	
Dimensions (diameter/height)	32/33.7	m
Design pressure/temperature	0.35/140	MPa/°C
Design leakage rate	0.5	vol%/day
Is secondary containment provided?	Yes, reactor building	

Reactor auxiliary systems

Reactor water cleanup,	capacity	22	kg/s
	filter type	Disk-type precoat filter	
Residual heat removal,	capacity	420	kg/s
	heat removal	24	MW
Coolant injection,	at high pressure	-	kg/s
	at low pressure	420	kg/s

Power supply systems

Generator transformer,	rated voltage	400	kV
	rated capacity	1300	MVA
Auxiliary power transformers,	rated voltage	10	kV
	rated capacity	-	MVA
Startup transformer	rated voltage	10	kV
	rated capacity	-	MVA
Medium-voltage busbars (6 kV or 10 kV)		2	
Number of low-voltage busbar systems		-	
Standby diesel generating units: number		2	
	rated power	6.8 (2×3.4)	MW
Number of diesel-backed busbar systems		2	
Voltage level of these		6	kV AC
Number of DC distributions		4	
Voltage level of these		220	V DC
Number of battery-backed busbar systems		4	
Voltage level of these		220	V AC

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	Axial, double-flow, condensing	
Number of turbine sections per unit (e.g. HP/LP/LP)	HP/LP/LP/LP	
Turbine speed	1500	rpm
Overall length of turbine unit	-	m
Overall width of turbine unit	-	m
HP inlet pressure/temperature	7.3/289	MPa/°C

Generator

Type	3-phase, turbo-generator	
Rated power	1445	MVA
Active power	1300	MW
Voltage	-	kV
Frequency	50	Hz
Total generator mass, including exciter	-	t
Overall length of generator	-	m

Condenser

Type	Shell-type	
Number of tubes	-	
Heat transfer area	-	m^2
Circulating water flow rate	51	m^3/s
Circulating water temperature	8	°C
Condenser pressure	4	kPa

Condensate pumps

Number	2	
Flow rate	-	kg/s
Pump head	-	MPa
Temperature	-	°C
Pump speed	-	rpm

Condensate polishing system

Full flow/part flow	1082/	kg/s
Filter type	-	

Feedwater tank

Volume	-	m^3
Pressure/temperature	-	MPa/ °C

Feedwater pumps

Number	$2 \times 70\%$	
Flow rate	-	kg/s
Pump head	-	MPa
Pump power requirement	-	MW
Feedwater temperature (final)	220	°C
Pump speed	-	rpm

Feedwater heaters

Number of heating stages,	low pressure	4
	high pressure	2
	feedwater tank	-

4.12.9 Measures to enhance economy and maintainability

The overall objective was to develop an economically competitive plant while achieving higher levels of safety than the existing fleet of commercial nuclear power plants and reducing or eliminating the risks associated with licensing and construction of a new nuclear power plant.

Economic competitiveness

Nuclear power plants can only be economically competitive if their power generating costs (i.e. capital cost plus operating, maintenance and decommissioning costs) are not higher than those of fossil-fired power plants (i.e. coal- or gas-fired units).

The dominant cost factors to be considered in connection with nuclear power plants are the amount of capital and the length of time the capital is not producing income; i.e. the construction and licensing period. The SWR 1000 design addresses all of these factors and offers a licensable design.

By designing the plant to rely on passive and active systems for safety functions a significant reduction in total plant cost has been realized. Active systems also require many more support systems than passive systems.

Savings can also be achieved by combining system functions, resulting in fewer systems. The SWR 1000 has eliminated some of the previously standard BWR systems by incorporating their functions into other systems, thereby reducing the total number of systems.

Maximizing plant availability is another technique that increases economic competitiveness. The SWR 1000 design achieves high availability through application of a wide range of experience gained from plants currently in service.

High level of safety

The high safety standard of current nuclear power plants is based on a complex system of redundant active safety equipment and all of the support systems required for the functioning of these systems. Achieving this safety standard entails high capital costs and considerable expenditure for operation and maintenance in terms of both personnel and equipment.

The SWR 1000 design achieves very high levels of safety while reducing the complexity of the safety systems through the use of “passive” safety equipment. As a result of this approach, the following attributes were realized:

- Clear and simple systems engineering
- Increased safety margins
- Slower reaction to off-normal conditions
- Increased grace periods (up to several days) after the onset of accident conditions before intervention by operating personnel is required
- Effect of human error on reactor safety is minimized or avoided entirely
- Much lower probabilities of occurrence of accidents leading to core melt
- The effects of a core melt accident are limited to the plant itself.

Reduction of risks associated with licensing and construction

All new passive systems have been tested either in full scale or in a scaled configuration and will be tested again with full-scale, prototype components.

A European utility group successfully assessed compliance with the European Utility Requirements. FRAMATOME ANP has also initiated the process of obtaining a 10CFR Part 52 Design Certification from the US NRC for the SWR 1000 to eliminate the risks inherent in the 10CFR Part 50 process for obtaining an operating license.

The construction time for the SWR 1000 is greatly reduced as a direct result of the simplification of systems and the incorporation of passive equipment. These two aspects of the design have resulted in fewer total components and have significantly reduced the number of safety-classified components.

4.12.10 Project status and planned schedule

The contract for the basic design extension phase between FRAMATOME and the German utilities expired in the fall of 2002. The following status was achieved:

- Basic design of nuclear island has been completed;
- Level 1 PSA (probabilistic safety analysis) is available;
- New passive safety systems have been thoroughly tested;
- Ambitious design objectives regarding safety and economics have been met;
- Potential developed for short construction period of 48 months;
- Assessment of compliance with the European Utility Requirements (EUR) has been completed.

An uprated SWR 1000 with a net electric output of approximately 1250 MW has been offered for the fifth nuclear unit to be built in Finland. Additionally, the process for obtaining Design Certification from the U.S. NRC was initiated in 2002.

4.13 WWER-1000/V-392 (ATOMENERGOPROJECT/GIDROPRESS, RUSSIAN FEDERATION)

4.13.1 Introduction

The principle of ensuring the safety of the personnel, the population and the environment against radiation effects exceeding the prescribed radiation doses is used as the basis for design NPP-92. This principle also addresses the standards for releases of radioactive substances and their content in the environment under normal operation conditions, anticipated operational occurrences, and in design and beyond-design-basis accidents during the plant service life. The objective of the design of the reactor plant and of the nuclear plant process systems was to achieve that the estimated probability of a severe fuel damage does not exceed 1.0E-5 per reactor-year and that the probability of accidental radioactive releases does not exceed 1.0E-7 per reactor-year. These values are currently prescribed by the Russian safety standards.

Nuclear power plant (NPP) safety is achieved by consistent implementation of the principle of "defence-in-depth" based on the application of a system of barriers on the path of spreading ionizing radiation and radioactive substances into the environment, as well as on a system of engineered safety and organizational provisions for protection of these barriers. A consistent implementation of the "defence-in-depth" principle is provided with the following:

- Installation of successive physical barriers on the path of spreading radioactive substances: fuel matrix, fuel element cladding, primary circuit boundary, containment;
- Taking into account postulated initial events that could lead to a loss of efficiency of those barriers;
- Determination, for each postulated event, of design measures and actions of operating personnel required to keep the integrity of the barriers mentioned, and mitigation of consequences of damage to such barriers;
- Minimization of the probability of accidents resulting in a release of radioactive substances;
- Consideration of beyond design basis accident management.

The design is developed in accordance with the latest versions of the Russian safety regulations for NPPs [1,2] by Atomenergoproject (Moscow) and EDO "Gidropress" under the scientific leadership of the Russian National Research Centre "Kurchatov Institute". The QA requirements of the IAEA and international standards ISO 9000 are taken into account in the design. The principal technical decisions of WWER-type reactors are proven by the WWER operational experience during about 1000 reactor-years including more than 300 reactor-years of WWER-1000 operation.

In the WWER-1000/V-392 plant safety concept, modern worldwide trends in NPP safety improvements are considered in order to meet the normative requirements for NPP safety, which are constantly becoming more strict, for as long a period as possible. The principal features that largely determine nuclear plant safety are as follows:

- Possibility of providing subcriticality with solid control rods at any moment of the plant life at coolant temperature below 100 °C;
- Application of horizontal steam generators with a large water inventory and with better conditions for natural circulation in the primary circuit in comparison with vertical steam generators;
- Application of an emergency core cooling system, based on the principles of both passive and active operation, that provides for the possibility of long term residual heat removal after accidents with primary leaks accompanied by a station blackout;
- Application of a passive core flooding system;
- Application of a quick boron supply system;
- Application of a double wall concrete containment;
- Application of a diagnosis system for periodical inspection of the safety important equipment (both during shutdown and on-line during reactor operation);
- Application of an automatic control system of improved reliability with self-diagnosis, and of an expert system giving advice to the operator;

- Application of an emergency system for discharging and purification of radioactive materials of the steam-gas mixture vented from the containment if the pressure exceeds the allowable values in beyond design basis accidents.

The NPP-92 design is being realized for Novovoronezh-2 NPP in Russia and NPP Kudankulam in India which are now under construction. Detailed analyses of NPP-92 design were performed by the organizations of Western Europe under the programme “Review of AES 92/EUR Compliance Analysis Report”.

4.13.2 Description of the nuclear systems

4.13.2.1 Primary circuit and its main characteristics

A schematic drawing of the reactor plant and the major plant systems is shown in Figure 4.13-1. The reactor plant includes a reactor coolant system, a primary pressure control system and a primary overpressure protection system. The reactor coolant system consists of 4 loops, with a horizontal steam generator and a reactor coolant pump in each one.

The primary pressure control and overpressure protection systems include a pressurizer, pressurizer spray valves, a bubbler, safety valves, pressurizer steam-gas mixture removal valves and throttling device, and connecting pipelines. The steam and water volumes of the pressurizer are determined in such a way that the set points of the safety valves will not be reached in any design basis condition (including design basis accident situation), even if the pressurizer spray fails to function. The discharge from the safety valves goes via a pipeline into the bubbler. The steam-gas mixture removal line with valves and throttling device connects the pressurizer steam volume with the bubbler; this line operates during plant startup when the primary coolant is heated, and in some accident situations.

The bubbler is a horizontal cylindrical vessel with elliptic bottoms filled with water to 2/3 of its volume. Two safety membranes are installed in its cylindrical part. Steam discharge headers with nozzles and a heat exchanger are arranged inside the bubbler.

There are three pressurizer spray lines in parallel; two of them which have an inner diameter of 125 mm, are equipped with shut-off valves, whereas the third, with a diameter of 105 mm, is provided with control valves. The pressurizer spray lines are also connected to the primary loop makeup water pump line. In the event of a primary system pressure increase beyond the operation set point of the pressurizer spray, primary coolant from the cold leg of the primary loop is injected through the pressurizer spray line and the nozzles in the pressurizer steam volume. The spray water can also be provided by the primary loop makeup water pumps in case the main coolant pumps have stopped operating.

4.13.2.2 Reactor core and fuel design

The WWER-1000 concept is based on an evolutionary approach, and the core design reflects the operation experience of current reactors. The core is designed to have negative reactivity feedbacks under all operating conditions to meet the basic safety objectives. Reactivity control is accomplished by changing of the boron concentration in the primary coolant and by moving control rod assemblies. As a rule, slow reactivity changes due to changes in burnup and amount of xenon are compensated by changes of the boron concentration, while rapid reactivity changes for adaptation of the power level to the load are accommodated by control rod insertions or withdrawals.

The power level of the reactor is monitored by means of ex-core instrumentation, but the 3-D power distribution in the core is determined by an in-core instrumentation system based on self-powered measuring detectors.

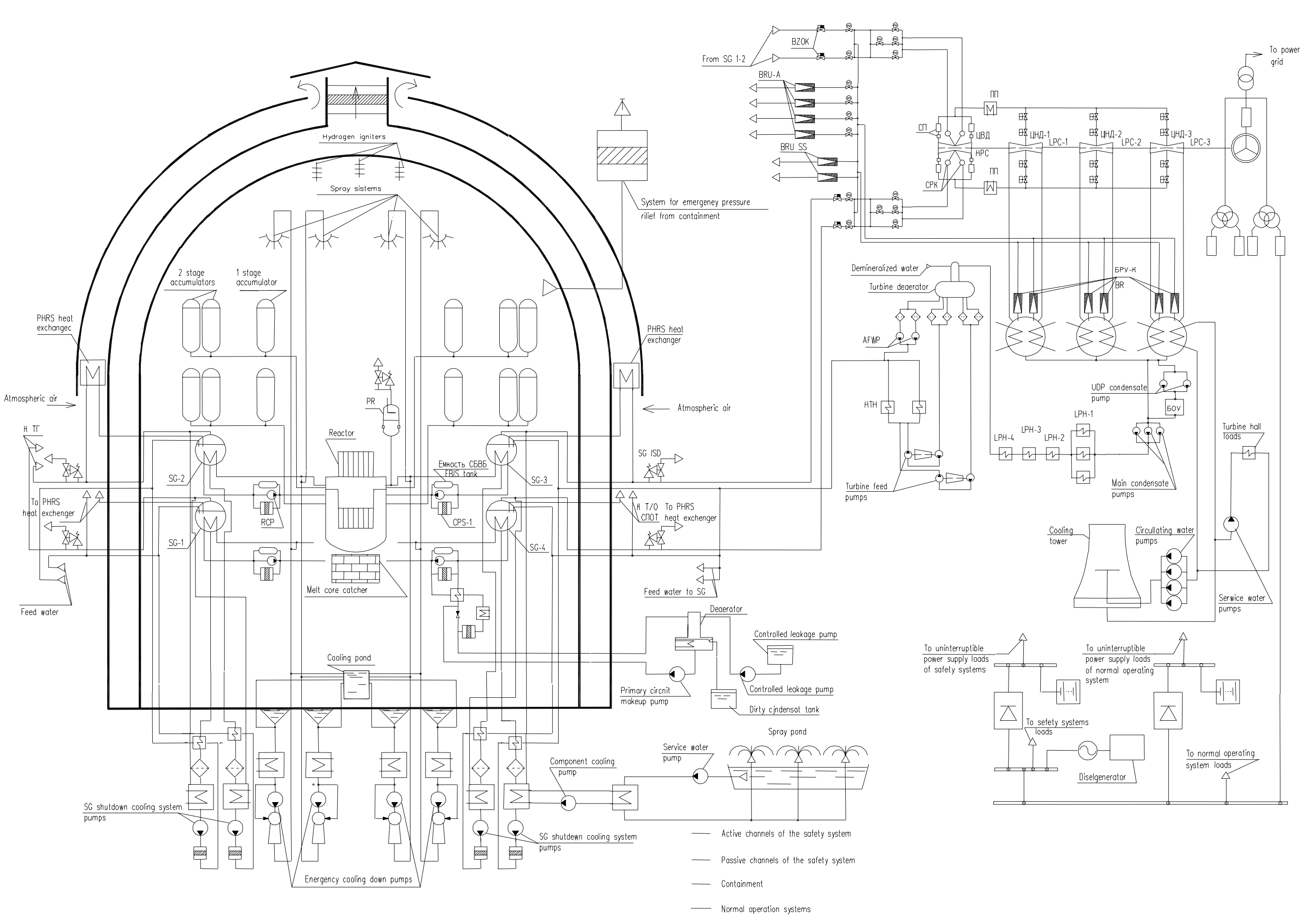


FIG. 4.13-1. NPP-92 design. Principal diagram.

The reactor core consists of 163 fuel assemblies, their active length is 3.53 m. Each fuel assembly comprises 311 fuel rods and 18 guide tubes. The fuel cladding is made of zirconium alloy tubing which contains sintered UO_2 pellets with an initial enrichment that does not exceed 4-4.4 weight %. The average linear heat rate of the fuel rods amounts to 16.67 kW/m. There are 121 control rod clusters which are also used by the reactor scram system for rapid shutdown of the reactor. Pitch electromagnetic drives with position indicators are used as driving devices for the control rod clusters. The drives are installed on the reactor vessel head.

The effective operation time between refuellings is 7000 effective full power hours. The average burnup of the discharged fuel is up to 43 MWd/kgU. The number of fresh assemblies loaded during annual refuelling is 54.

4.13.2.3 Fuel handling and transfer systems

The fuel handling and transfer systems are intended for loading of fresh fuel assemblies and replacing spent fuel assemblies, shutdown absorber rods and burnable poison absorber rods. Fuel handling and transfer are performed under water for radiation protection. For fuel handling to take place, the reactor is shut down and depressurized, the head of the reactor pressure vessel removed and the control rod guide tube unit extracted from the reactor vessel, and the concrete well above the reactor and the fuel pool filled with water.

The fuel handling and transfer systems include the following compartments with necessary equipment: the reactor concrete well; the fuel pool; and the transfer compartment. The fuel pool is located in the vicinity of the reactor concrete well; they are connected by a transfer channel designed for transportation of one fuel assembly at a time. The fuel pool is provided with storage structures for spent fuel assemblies; these structures consist of separate sections designed for storing of fuel assemblies and sealed containers for failed fuel assemblies. The transfer compartment is provided with a universal nest for location of a plant-internal transportation unit for fresh fuel assemblies and for a transport cask for spent and decayed fuel assemblies. The transfer compartment is connected to the fuel pool by the transfer channel through which loading of fresh fuel assemblies take place. The fresh fuel assemblies are taken out of the plant-internal transportation unit by the refuelling machine and installed in the core in accordance with the core refuelling chart.

After unloading from the core, the spent fuel assembly is placed in a container with a fuel assembly defect monitoring system, and then transported to the storage structure or the sealed container for spent and decayed fuel assemblies in the fuel pool depending on the results of the defect inspection. Manipulations for control rod assembly replacements are similar to those for fuel handling; control rod clusters and burnable poison rods extracted from the reactor core are installed in empty fuel assemblies or in storage structures in the fuel pool for storage. The refuelling machine handles only one fuel assembly, one control rod cluster, or one fuel assembly with the control rod cluster inside it, at a time. Heat removal from the reloaded fuel assemblies is accomplished by the fuel pool cooling system.

4.13.2.4 Primary components

Reactor pressure vessel

A schematic drawing of the reactor assembly is shown in Figure 4.13-2. The reactor vessel is similar to that of a serial WWER-1000 reactor.

The reactor vessel consists of a flange, upper and lower shells of nozzle zone, support shell, cylindrical shell and elliptic bottom, welded together by circumferential seams. The vessel has two rows of nozzles of 850 mm inner diameter (four nozzles in each row); there are two nozzles for the pipeline connections of the ECCS hydroaccumulators on each row level, and there is one instrumentation penetration nozzle in the upper row. The inner surfaces of the vessel and the nozzles are plated with a corrosion-resistant layer. A separation ring is welded to the vessel inside between the upper and lower nozzle rows for separation between the inlet and outlet coolant flows, and a support rib is made on the vessel support shell for vessel attachment on the support

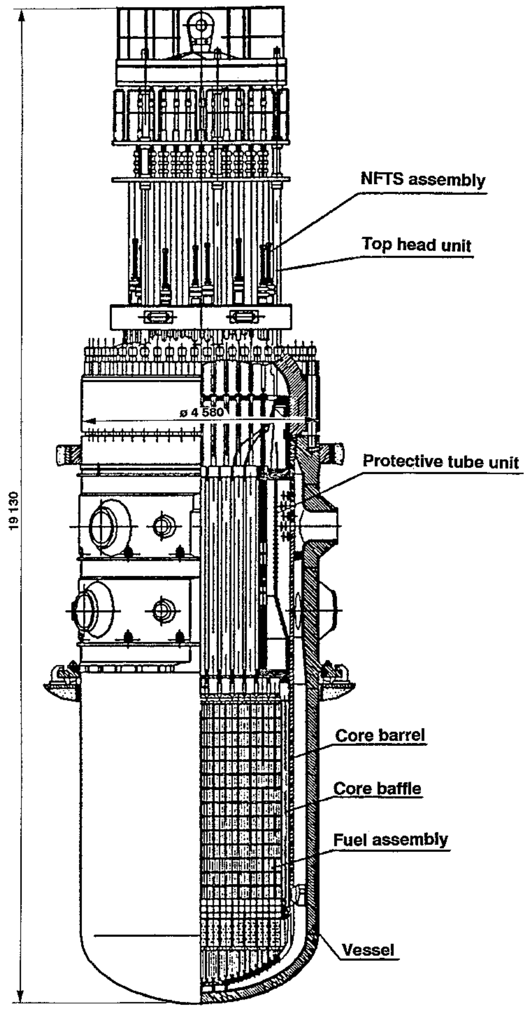


FIG. 4.13-2. V-392 reactor assembly.

structure. The vessel cover consists of a truncated ellipsoid and a flange that are connected with each other by a circumferential weld seam. Penetrations for in-vessel instrumentation, nozzles for gas removal lines, and holes for control rod housings are provided on the cover ellipsoid.

Reactor internals

Structures providing support and positioning of the core and control rods are installed inside the reactor vessel; the internal structures including the core support structures, outer core structure and upper protective tube structure also serve as coolant low guides for the core heat removal.

Steam generators

The steam generator (SG) is of the horizontal, single-vessel type, with an immersed heat exchange area consisting of tube bundles arranged horizontally. The SG is a modernization of the standard SG PGV-1000. The operation experience from operating WWER-1000 and WWER-440 SGs has been taken into account when designing the SG for WWER-1000/V-392 reactor plant. In particular, the perforated part of the primary collector is made of stainless steel 0KH18N10T that has shown good properties during the operation of the WWER-440 SG primary collectors. For internals inspection, hatches of 500 mm diameter on the elliptic bottom, as well as hatches of 1000 mm diameter in the cylindrical part of the steam generator, are provided.

Pressurizer

The standard WWER-1000 pressurizer is used for the V-392 plant. The pressurizer with primary pressurization system is designed for maintaining primary system pressure within an acceptable range in all reactor plant conditions. The pressurizer is a vertical vessel mounted on a cylindrical support structure. There are nozzles for spray pipelines and a safety valve steam discharge line on the upper end, and a nozzle for connection between the pressurizer and the hot leg is provided in the vessel bottom. Nozzles for level gauges and casings for thermo-resistors measuring coolant temperature inside the vessel are located on its cylindrical shell. A spray device and a tubular electric heater are located inside the pressurizer. The spray device is intended for water spraying in the steam volume and condensation of steam; it is made as a discharging header fastened to the top of the vessel. The inside surface of the pressurizer is plated with a corrosion resistant material. All pressurizer internals are made of austenitic stainless steel.

Reactor coolant pumps

The reactor coolant pump (RCP) is a modernization of the pumps proven by successful operation at serial WWER-1000/V-320 reactor plants. The RCP is a vertical, single-stage, centrifugal pump with an autonomous lubrication system housed in a spherical case. The RCP sub-systems prevent the escape of radioactive coolant out of the primary system. The electrical motor is of the vertical type, three-phase, and with two speeds. A non-combustible lubricant is used in the electrical motor. The shaft prevents the coolant leak in case of loss of power for 24 hours and loss of sealing water and other cooling media.

Main coolant lines

The reactor pressure vessel nozzles for the coolant inlet and outlet (the cold and hot legs) have the inner diameter of 850 mm.

4.13.2.5 Reactor auxiliary systems

Process systems and special water treatment (SWT) building plan dimensions are 45.0 x 66.0 m (Safety class II). This building is connected to the containment building and adjoins the latter by the long wall from the side of transportation trestle. The building houses auxiliary primary systems including special water treatment unit. The building is made of monolithic reinforced concrete.

Special sewage, borated and service water drainages collection systems are located at lower elevations. Laboratories are designed at elevation 7.2 m. The I&C, control and protection system premises, plenum ventilation centre, exhaust ventilation centre and stack are located at higher elevations. Access lock into the containment leaktight space is at elevation 31.8 m; entrance into the building and exit out of the building is through sanitary-social service building.

4.13.2.6 Operating characteristics

The reactor plant is four-loop one, primary coolant temperature at the reactor outlet is 321^{0}C, design primary pressure is 17.6 Mpa. One RCP with external motor with necessary inertial performances and one horizontal steam generator with submerged heat exchanging surface are installed in each loop; live steam pressure is 6.37 MPa (design pressure is 8,0 MPa); steam capacity of the reactor plant as a whole is about 5900 t/h.

4.13.3 Description of the turbine generator plant systems

Turbine hall, deaerator rack and oil handling building depend mainly upon the type of turbine plant with condensers located in the basement, three low pressure cylinders and water cooled generator as well as upon the layout of auxiliary systems and equipment. The turbine building and deaerator rack, including oil handling building belong to Safety class II. The turbine hall plan dimensions are 36.0 x 102.0 m, its height is 40.8 m. The turbine hall carcass is designed to be made of steel structures. The arrangement of turbine hall with its end face towards reactor department enables to use the layout volume to the best for arrangement of equipment and locate turbine steam exhausts as close to the reactor building as possible. The turbine plant foundation is provided with vibration dampers so the transfer of dynamic and vibrational effects upon civil srtuctures of platforms and ceilings resting on turbine plant foundation pillars and by lower support plate are practically avoided.

4.13.3.1 Turbine generator plant

No information provided.

4.13.3.2 Condensate and feedwater systems

No information provided.

4.13.3.3 Auxiliary systems

No information provided.

4.13.4 Instrumentation and control systems

4.13.4.1 Design concept, including control room

The reactor plant control system secures fulfillment of the following main functions:

- Monitoring of the unit operation, radiological situation, state of equipment and systems in all conditions;
- Remote control;
- Automatic control of reactor plant, secondary circuit and auxiliary systems parameters;
- Process protection and interlocking;
- Emergency and preventive protection of the reactor.

The following priority of control commands (in the order of decreasing priority) is secured in the control system of the reactor plant:

- Emergency and preventive reactor protection and control of safety systems;
- Process protection and interlocking;
- Manual remote control;
- Automatic control (the main controlled parameters are: neutron flux in the core, primary pressure, secondary pressure, water levels in steam generators and the pressurizer);
- Recording and archiving of the main parameters under normal and emergency conditions.

In case of a main control room (MCR) failure, for example during a fire, the reserve control room (RCR) is used to provide:

- reactor shutdown,
- monitoring of subcriticality,
- reactor cooldown,
- putting into operation of confining systems.

Possibility of control of the systems important for safety is retained from RCR. Ensurance of habitability under loss of regular ventilation systems during a safe shutdown earthquake (SSE) and associated fire, or other destructions on the site, is provided for the reserve control room. Local control panels which do not require interaction with the MCR and the RCR are provided. Their existence, in a number of cases, is determined by considerations of NPP layout. The RCR is provided with an access admittance check system.

4.13.4.2 Reactor protection and other safety systems

The following systems belong to the class of automatic safety systems:

- reactor emergency protection system,
- primary overpressure protection system,
- emergency core cooling system,
- system of passive heat removal from the reactor plant,
- system of quick-acting isolation valves in steamlines,
- secondary overpressure protection system,
- quick-acting boron supply system,
- emergency diesel-generators,
- emergency system of reliable direct and alternate electric current power supply.

Reactor emergency protection system

The reactor emergency protection system provides reliable disconnection of electric power supply and, as a consequence, a drop of emergency protection rods into the core. In this case, disappearance of signal of original cause does not stop the initial action of the emergency protection. The solid rods of the emergency protection actuate in response to the following signals:

- decrease of reactor period,
- increase of neutron flux,
- decrease of margin to saturation temperature in any hot leg,
- increase of coolant temperature in any hot leg,
- decrease of pressure differential over the primary coolant pumps,
- de-energization of several primary coolant pumps,
- decrease of pressure in the reactor,
- increase of pressure in the reactor,
- increase of pressure in a SG,
- decrease of pressure in a SG coinciding with a definite increase of the primary and secondary saturation temperature difference,
- decrease of water level in a SG,
- increase of pressure in the containment.

The parameters chosen permit to secure the necessary reduction of reactor power for meeting the design criteria under all design conditions. Automatic disconnection of power governors, and interlocking of all operator's actions on control rods occur when the emergency protection operates.

Two sets of instrumentation are provided, generating the commands for the emergency protection and operating in parallel using an "or" logic. The signals for operation of the emergency protection are generated using a "2-out-of-3" majority logic in any set. However, with the aim of enhancing NPP safety, failure of the emergency protection system of the reactor is postulated in some beyond design basis accidents by considering scram failure under operational occurrences and in design accidents.

4.13.5 Electrical systems

4.13.5.1 Operational power supply systems

The high voltage switchgears 500 kV and 220 kV are provided in the design for NPP power output into the grid. Outdoor switchgear 500 kV is constructed in 500 kV line. Metal-clad SF_6 gas isolated switchgear (MCS) is constucted in 220 kV line. Application of MCS-220 kV instead of outdoor switchgear is caused by its vicinity to the cooling tower because of terrain relief.

According to the design each Unit shall be equipped with one completely water cooled turbine generator of 1100 MW. Generator and main transformer are connected by shielded busducts with generator breaker between them which is capable to disconnect the short circuit current. Two working transformers of 63 MVA each are installed in the tap between the generator breaker and main transformer.

Availability of generator breaker enables schedule Unit startup and shutdown from the grid through main transformer and to provide auxiliary power supply from working auxiliary transformers in case of generator or process part of the Unit failure without change-over to standby transformer and so enhances the Unit reliability considerably.

The NPP auxiliary power supply system contains the sources of working, standby and emergency power supply. Auxiliary power supplies are divided into off-site and internal ones. The power grid with its power plants in the off-site power supply. Off-site power may be supplied to NPP auxiliaries through working auxiliary transformes or through standby transformers 220/6.3-6.3 kV. Internal normal operation auxiliary power sources are the turbine generators and emergency auxiliary power sources-diesel-generators and storage batteries.

The NPP auxiliary power system is designed to supply loads supporting:

- NPP normal operation;
- Unit bringing into safe condition and maintaining it so under normal and emergency operation conditions;
- Reactor plant state 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 Unit normal operation. The design envisages four normal operation sections of 6 kV – in accordance with the number of RCPs. Power to each RCP is supplied from individual section so that more stable operation of the Unit is provided in case of loss of 6 kV sections as loss of RCP requires Unit power reduction or its shutdown.

4.13.5.2 Safety-related systems

Each independent subsystem consists of two trains with mutual redundancy allowed. Installation of diesel-generator 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 loss of normal power supply.

Emergency power supply system storage batteries are intended to:

- One storage battery is to provide power to control and automation and relay protection devices of emergency power supply system elements, as well as for emergency lighting of loads of this channel of the emergency power supply system. Time of the battery descharge is 2 hours;
- The second battery is to provide power to I&C hardware. Duration of battery discharge is also 2 hours;
- The third battery is to provide power to reactor control and monitoring devices in case of total loss of a.c. power. Time of battery discharge is 24 hours.

Emergency power supply system electric equipment is located in the standby diesel power station (SDPS). The SDPS for each subsystem is arranged in two separate buildings. Each building consists of two physically separated cells. Each cell houses equipment of one train; the trains are separated by structures of not less than 1.5 hour fire resistance limit.

4.13.6 Safety concept

4.13.6.1 Safety requirements and design philosophy

The design solutions on safety for nuclear power plants with WWER-1000 reactors (V-392) are aimed at developing a NPP with an enhanced level of safety so that the total risk associated with NPP operation is as low as can reasonably be achieved. The development of the safety concept is based on the detailed analyses of the experience and the PSA results of operating NPP with V-320 reactors and on the requirements of Russian regulatory documents as well as the recommendation of IAEA documents on safety. The following main deterministic principles are included in the safety concept:

- single failure principle,
- diversity,
- physical separation,
- defence from common cause failure,
- defence from human errors,
- application of passive safety systems,
- protection from internal and external hazards,
- assurance of more high reliability level for performance of safety functions with high frequencies of demands,
- decreasing the frequency values of catastrophic failures to negligible levels.

According to the Russian safety standards, probabilistic safety goals have been established for WWER-1000/V-392 design (core damage frequency less 1.0E-5 per reactor year and frequency of large emergency release less 1.0E-7 per reactor year). Based on these requirements, the following decisions were used in NPP-92 design:

1. The mutually redundant systems of passive and active operation principles or systems with diverse design of components are used for performing the main safety functions:
    - An upgraded emergency protection system with the number of control rods two times more than that used in the V-320 reactor plant and a quick boron injection system to bring the reactor in a subcritical state and maintain it in this state over a wide range of operating parameters (the emergency protection system is capable to maintain the subcritical state up to a temperature below 100^0C).
    - The active and passive systems for emergency heat removal through the secondary coolant circuit. Both these systems can remove heat during an infinite time, whereas the emergency heat removal system for NPPs with a V-320 reactor can operate only for a limited time (about 30–40 hours), which is determined by the inventory of coolant in its tanks.
    - The active emergency core cooling system (ECCS) and the 1st and 2nd stage hydroaccumulators to maintain the inventory of reactor coolant in the core in cases involving leaks from the reactor coolant system. The 2nd stage hydroaccumulators together with the 1st stage hydroaccumulators provide a redundancy to the active ECCS in terms of the function of maintaining the inventory of coolant in the core during 24 hours after the accident. This time can be used to restore the operability of the active ECCS in case of its failure.
2. The individual trains of active safety systems (the emergency cooldown system and ECCS) can be used to perform functions of normal operation. Here, most of the components of these systems are in the states that are similar to those when the required emergency functions are performed. Using such operating modes of these systems, it is possible to enhance their availability indices and to provide additional protection against common cause failures.

3. To ensure the deep defence from the operator errors. This is based on using the passive safety systems which do not require the operator actions for their operation and on using the high level of automatization in active safety systems.

4. Application of leak before break concept to decrease the frequency values of large LOCAs and large leakages from the primary to the secondary circuit up to a negligible level.

5. To develop the containment system for the mitigation of radioactive releases at severe accidents with core melt. The double containment system with passive hydrogen removal system, ventilation and radioactive purification system of volume between internal and external containments and catch for melting core is used in NV NPP-2 design.

4.13.6.2 Safety systems and features (active, passive and inherent)

The safety systems consist of the emergency core cooling system, passive residual heat removal system, active residual heat removal system, quick boron supply system, primary overpressure protection system, secondary overpressure protection system, system of quick-acting isolation valves in steam lines. The overall configuration and functions of the safety systems are shown in Figures 4.13-1 and 4.13-3 through 4.13-6.

Emergency core cooling system

The emergency core cooling system (see Fig. 4.13-4) has been designed to provide a possibility of longterm residual heat removal in case of primary leak accidents accompanied by a station blackout. The ECCS involves two subsystems:

- A passive subsystem using hydroaccumulators (HA) with nitrogen under pressure (1st stage HA);
- A passive core flooding subsystem (2nd stage HA);
- An active subsystem with high pressure injection pumps.

The hydroaccumulators with nitrogen under pressure will provide the coolant injection during the first stage of such an accident, and the active subsystem takes over when the hydroaccumulators have been emptied.

The passive core flooding subsystem includes four groups of hydroaccumulators under atmospheric pressure (2nd stage HA) which are coupled with the pipelines connecting the ECCS hydroaccumulators with the reactor. The hydro-tanks of the passive core flooding system are connected to the primary system at 1.5 MPa and allow to flood the core due to the hydrostatic pressure of the water column, and to remove the reactor residual heat in the last stage of a LOCA for at least 24 hours.

The active part of the ECCS includes two independent trains having an overall redundancy within each train. Each of the 4 subtrains thus formed is capable to fulfill the necessary system functions. Each subtrain includes the sump of the containment, a high pressure injection (HPI) pump, a jet pump installed on the discharge side of the HPI pump, an emergency cooling-down heat exchanger and pipelines and fittings. The energy supply for the active elements of the system is provided by the reliable emergency electric power supply system. Each of the four subtrains of the system has its own subtrain of reliable electric power supply, including a diesel-generator. One train of active ECCS is used for heat removal from spent fuel at normal operation condition.

Passive residual heat removal system

A passive residual heat removal system (see Fig. 4.13-5) is included in the design to remove heat from the reactor plant. The design basis of the PHRS is that in a case of a station blackout, including loss of emergency power supply, the

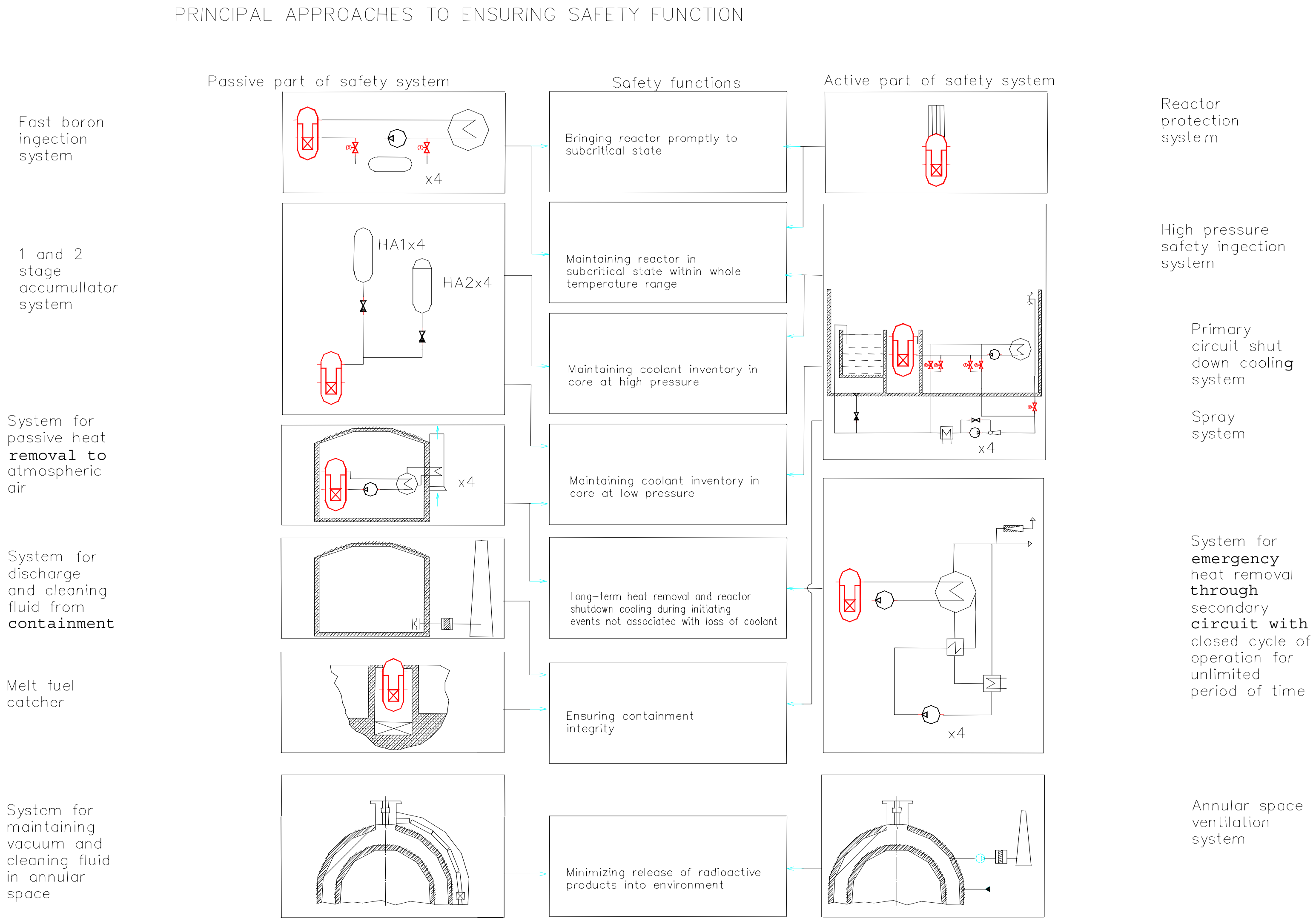


FIG. 4.13-3. Principal approches to ensuring safety function.

removal of residual heat should be provided without damage of the reactor core and the primary system boundary during unlimited time. The PHRS consists of four independent trains, each of them being connected to the respective loop of the reactor plant via the secondary side of the steam generator. Each train has pipelines for steam supply and removal of condensate, valves, and an air-cooled heat exchanger outside the containment. Part of the PHRS is an air-cooled heat exchanger that is installed outside the containment. The steam that is generated in the steam generators due to the heat released in the reactor, condenses and rejects its heat to the ambient air. The condensate is returned back to the steam generator. The motion of the cooling medium takes place in natural circulation.

Active residual heat removal system

The active emergency residual heat removal and cooling system (EHRS) through the secondary circuit consists of 4 close circuits of secondary coolant cooling - one per each steam generator (see Fig. 4.13-6). The EHRS has the structure of 4 x 100 %, and each train can perform the EHRS functions as a whole. Under normal operation conditions the said circuits are used for SG blowdown cooling.

Quick boron supply system

The quick boron supply system (QBSS), being developed as an additional reactor trip system, comprises a system of 4 special loops bypassing the main coolant pumps. Each loop consists of a hydroaccumulator containing concentrated boron acid solution, and of pipelines with quick-acting valves that do not require electric power for their opening. These valves open during occurrences and accidents with failure of scram, and concentrated boron solution is pressed out of the hydro-accumulators into the primary loops, and further into the reactor. In case of a station blackout the boron solution delivery occurs in the period of reactor coolant pump (RCP) coast-down. The RCPs have a considerable flywheel inertia which provides the possibility of ejecting all boron concentrate from the QBSS hydro-accumulators. The amount and concentration of the boron solution are chosen to provide a definite equivalency from the viewpoint of reactivity inserted by this system and by the solid absorber scram. In fact, this system, being part of the primary coolant circulation system, allows to consider a reactor plant with such a system as a plant with increased “inherent” safety.

Primary overpressure protection system

The system comprises three safety valves for discharging steam or a steam - water mixture from the pressurizer if its pressure increases above the permissible level, as well as a subsystem for receiving a steam - water mixture. This subsystem involves a bubbler and pipelines connecting it to the outlets of the safety valves.

Secondary overpressure protection system

The secondary overpressure protection system is intended for preventing the secondary pressure to increase above the permissible value. The system incorporates quick-acting steam dumping valves and steam generator safety valves.

System of quick-acting isolation valves in steam lines

Quick-acting isolation valves in the steam lines close at:

- Increase of water level in the SGs above the permissible level; and
- Increase of radioactivity in the SGs above the permissible level, on the appearance of signals of a steamline rupture.

They are intended, respectively, for the protection of the turbine from steam of high humidity, for preventing radioactivity releases from the SGs, and for restricting the steam blow down after a rupture of the secondary circuit.

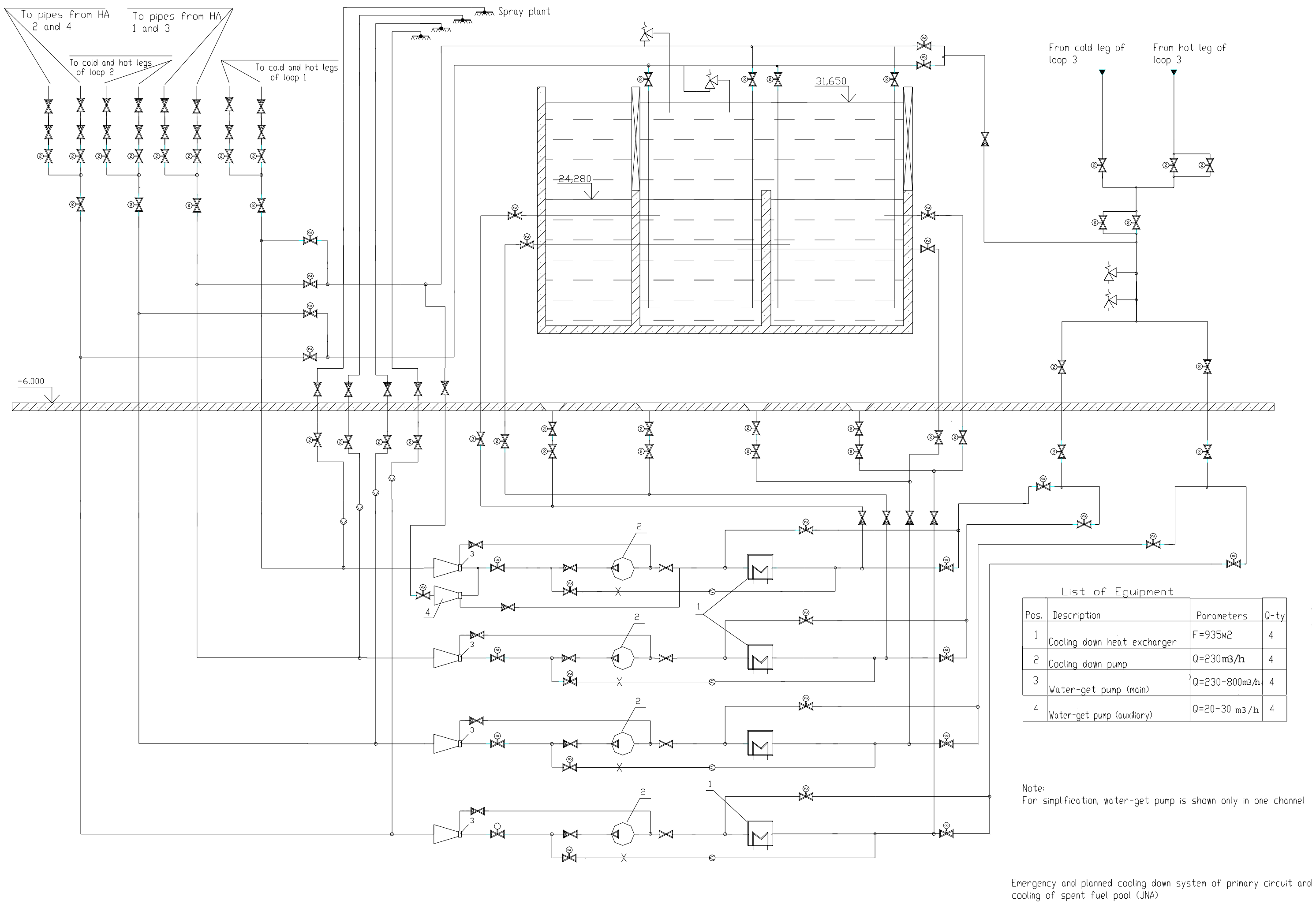


FIG. 4.13-4. Principal diagram of ECCS.

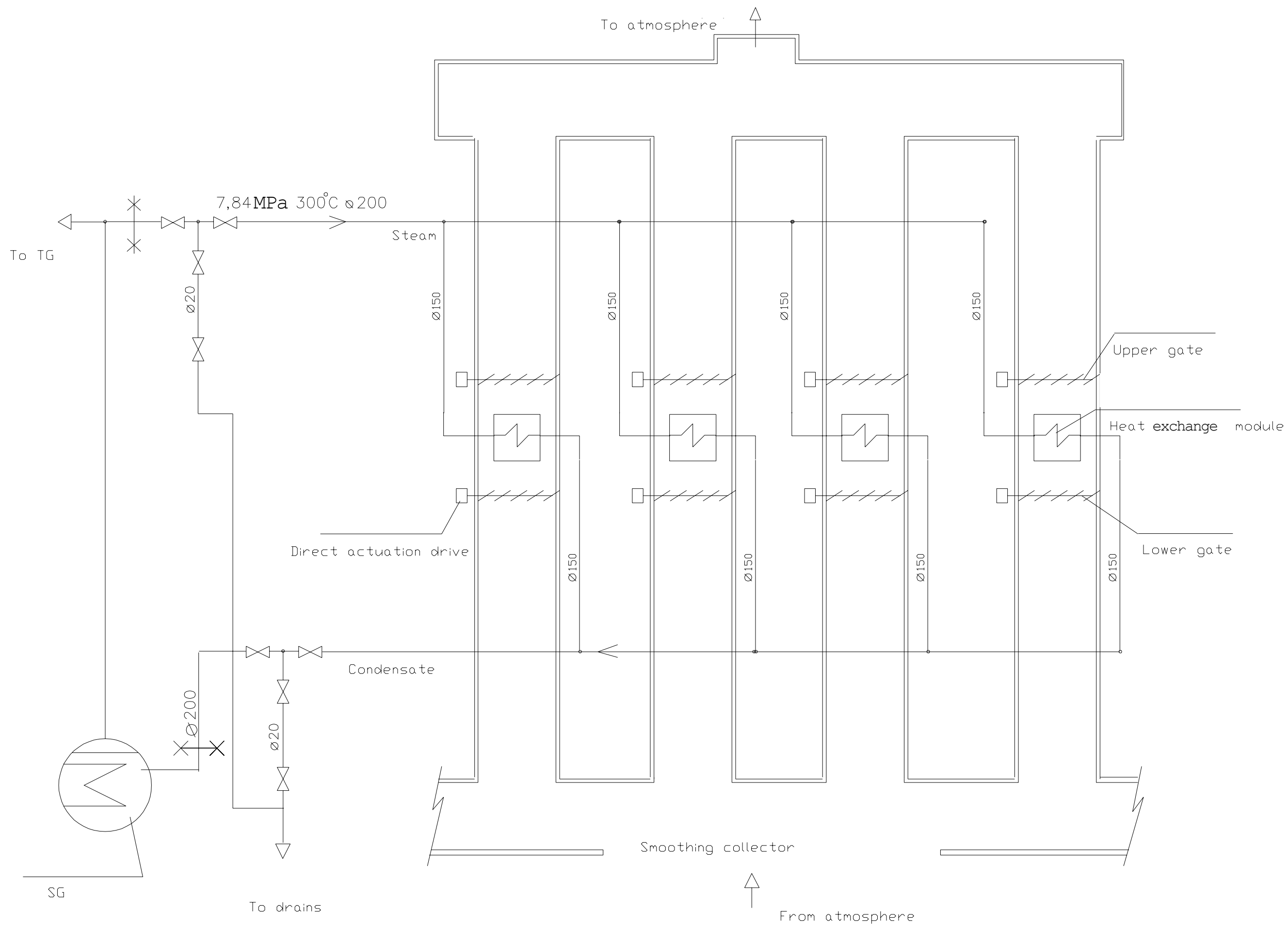
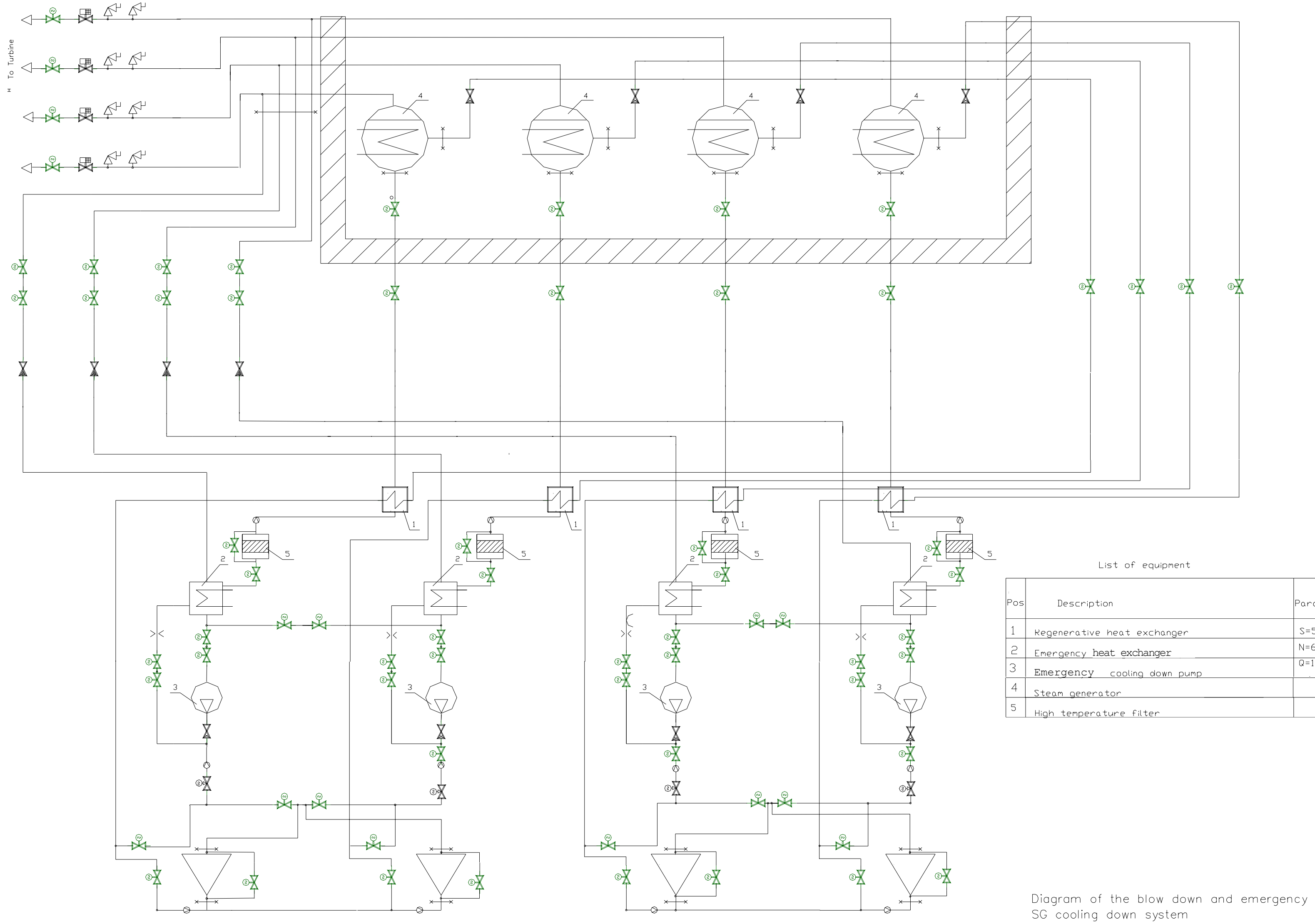


FIG. 4.13-5. Principal diagram of PHRS.



List of equipment

Pos	Description	Parameters	Q-ty
1	Regenerative heat exchanger	S=55 $м^2$	4
2	Emergency heat exchanger	N=64 MW	4
3	Emergency cooling down pump	Q=150m3/h	4
4	Steam generator		4
5	High temperature filter		4

FIG. 4.13-6. Principal diagram of EHRS.

4.13.6.3 Severe accidents (beyond design basis accidents)

Severe accident prevention and mitigation strategy is based on using the passive safety systems as follows:

- PHRS to prevent core damage for transient initiating events in case of EHRS failure and to remove the case of active ECCS failure;
- Passive core flooding system (2nd stage HA) to prevent core damage for LOCA's in case of active ECCS failure during a time period of at least 24 hours. This time is used to recover the operability of active ECCS;
- Double containment with passive hydrogen removal system, ventilation and purification system of volume between inner and outer contaiments and catcher for corium to mitigate the releases at severe accident with core melting.

Functional and structural diversity of safety systems as described above enables the deep protection against common-cause failures. Application of passive systems and active systems actuating without personnel interference enables the deep protection against human errors. These result in high safety level of WWER-1000/V-392 plant in terms of estimated frequency of severe core damage. The relevant PSA-1 data are shown in the Table 4.13-I demonstrating that the probabilistic safety goals are met with the essential margin.

TABLE 4.13-I. ESTIMATION OF CONTRIBUTION OF INITIATING EVENTS

Initiating event category	**Contribution to core damage frequency**	
	absolute, 1/year	relative, %
1. Internal IE during reactor power operation	2.6E-8	48
2. Internal IE under shutdown conditions	2.2E-8	40
3. Seismic effects	5.9E-9	11
4. Fires in the NPP premises	4.0E-10	1
All the categories of IE	5.4E-8	100

4.13.7 Plant layout

4.13.7.1 Buildings and structures, including plot plan

The main building layout is of a monoblock type, main and ancillary equipment of the reactor plant and turbine of each Unit is located in separate compartments. The main building which is a standardized module combines reactor and turbine departments and a sanitary-social building.

4.13.7.2 Reactor department

The reactor department is a number of adjacent compartments containing the reactor plant and systems supporting its normal operation and ensuring emergency shutdown of the unit.

The reactor department consists of the containment building, buildings of protection safety systems, monitoring and control systems building, normal operation process and the special water treatment systems building. All the buildings are constructed on separate foundation slabs. Proceeding from the condition of independent response under static and special dynamic effects the gaps 400 mm between buildings are provided.

4.13.7.3 Containment

The containment building consists of a cylindrical containment (leaktight part) and two adjacent buildings from opposite sides (non-leaktight part) located on the same foundation slab. The containment houses safety-related systems, so it is designed to withstand external effects and belongs to Safety class 1. The building plan dimensions are 73.2 x 43.2 m, building height is 89.4 m. The containment is an accident localization system and consists of two shells: inner leaktight containment and external one to protect the inner shell from the external effects. The reactor plant, spent fuel pool, ancillary process systems working at primary parameters, ventilation systems and equipment providing the fuel handling and repair operations are located under the containment.

The central part under the containment is occupied by the reactor. Located to both sides of the reactor pit are the spent fuel pool and internals inspection wells, two compartments housing steam generators, RCPs, main circulation pipelines, pressurizer, bubbler and quick boron supply system tanks. The RPV inspection machine and core catcher (for core melt accidents) are located on the slab of leaktight containment under the reactor. Pulse valves actuators maintenance rooms, purification systems rooms (SWT-1, SWT-2), contaminated pipelines valve control chambers, ventilation plants are located around the reactor pit. The ECCS tanks of the 1st and 2nd stages are located also inside the containment at at the maintenance elevation. The contaminated equipment washing unit is located next to the spent fuel pool from the side of the equipment lock.

For execution of transportation operations through the localization boundary the containment is equipped with locks designed to ensure tightness under design basis accidents and design external effects. Personnel access into the containment is through the main lock to the maintenance elevation from the control access area of the auxiliary process systems building. An emergency lock is provided at the lower elevation to ensure emergency exit of personnel from the containment. All fuel and equipment handling operations are carried out through the equipment lock at the maintenance elevation and trestle located outside the containment.

The containment building basement part is the space between the leaktight containment slab and building foundation slab. The height of the basement is defined by structural features of cooling down pumps. The containment basement houses the primary circuit and spent fuel pool cooling systems, primary circuit I&C and radiation monitoring assemblies (immediately under the core), intermediate circuit system, steam generators emergency cooling down and blowdown systems, I&C and radiation monitoring and I&C rooms of the systems located in the basement. In the building adjacent to the containment from the side opposite to the turbine building there are filters of containment overpressure release filters, service water emergency inventory tanks and exhaust ventilation center.

The systems located in the non-leaktight part of the containment building (except for the system of containment overpressure release) are divided into two independent channels with 100% redundancy in each channel. Rooms of different channels are separated by walls. The area of one channel (including rooms, corridors, staircases) is completely isolated from the area of another channel and no communication lines (pipelines, cable and ventilation ducts) of other channels pass through it. The corridors of the area of one channel are separated by a vestibule from the common corridor in order to provide the fire protection measures. Common corridor and emergency exits into the outdoor area are used to provide for the evacuation of personnel from those areas.

The outer containment is made as a cylinder with a spherical dome made of monolithic reinforced concrete with a cylindrical and dome walls thickness of 600 mm and an inner diameter of 50.8 m. The outer containment absorbs loads of external effects: hurricanes, tornadoes, external shock wave, aircraft crash. The inter-containment gap is 2.2 m due to conditions of maintenance of the inner containment prestressing system and accessibility of surfaces for visual inspection. The inter-containment gap enables gas-air media leaks controlled collection. The inner surface of the outer containment shall be provided with polymeric coating ensuring required tightness of the outer containment. Passive heat removal system heat exchangers are located on the outer containment. The heat exchangers layout on the outer containment is designed in such a way that heat exchangers of only one steam generator could be damaged in case of an aircraft crash.

4.13.7.4 Turbine building

The turbine hall, deaerator rack and oil handling buildings depend mainly upon the type of turbine plant with condensers located in the basement, three low pressure cylinders and water cooled generator as well as upon the layout of auxiliary systems and equipment. The turbine building and deaerator rack, including oil handling building belong to Safety class 2. Turbine hall plan dimensions are 36.0 x 102.0 m, its height is 40.8 m.

The turbine hall carcass is designed to be made of steel structures. The arrangement of the turbine hall with its end facing towards the reactor department enables to use the layout volume to the best for arrangement of equipment and locate turbine steam exhausts as close to the reactor building as possible. The turbine plant foundation is provided with vibration dampers so the transfer of dynamic and vibrational effects upon civil structures of platforms and ceilings resting on turbine plant foundation pillars and on the lower support plate are practically avoided.

4.13.7.5 Other buildings

Two protection safety system buildings, standard from the viewpoint of their layout, belong to Safety class 1 and are located on opposite sides of the monitoring and control systems building to prevent their simultaneous destruction due to 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 monitoring and control systems electrical equipment is located between the protection safety system buildings. It houses the premises of the main control room, control and protection system and information computer system panel. The above buildings are made of monolithic and prefabricated-monolithic reinforced concrete.

The standby diesel power station (SDPS) is intended to supply power to safety system loads under NPP blackout. Equipment of each safety system train is located in a respective isolated SDPS cell. Four cells with plan dimensions of 30.0 x 33.0 m and 16.0 m in height are provided for each unit. The SDPS cells are arranged in pairs, with their short sides adjacent to each other and are located in two buildings preventing their simultaneous destruction in case of an aircraft crash. Each cell houses the standby diesel power plant itself and an intermediate circuit and vital consumers service water supply pumphouse. The building is made of monolithic reinforced concrete structures.

Sheltered Centre of Emergency Actions Management at NPP (SEAMC) is designed in a sheltered, two-storey underground structure with plan dimensions of 24.0 x 54.0 m belonging to seismic category 1 and Safety class 1. The building houses the standby control rooms of Units 1 and 2 (SCR1 and SCR2)and NPP central control room (CCR) and provides the shelter for 900 persons. The SCR is intended to shutdown the unit in case of MCR failure. From SCR it is possible to monitor and initiate the safety systems and remove heat from the reactor plant. The reactor plant and spent fuel pool state can be monitored from SCR under all operating conditions including blackout. The SCR availability for 24 hours using storage batteries is ensured in case of loss of power. The system of crucial parameters recording ("black box"), ensuring information preservation in case of accident at the Unit, is located at SCR.

4.13.8 Technical data

General plant data

Power plant output, gross	1068 MWe
Power plant output, net	999.5 MWe
Reactor thermal output	3000 MWt
Power plant efficiency, net	33.3 %
Cooling water temperature	20 ^{o}C

Nuclear steam supply system

Number of coolant loops	4
Primary circuit volume, including pressuriser	396 m^3
Steam flow rate at nominal conditions	5880 t/h
Feedwater flow rate at nominal conditions	1648 kg/s
Steam temperature/pressure	278/6.27 ^{o}C/MPa
Feedwater temperature/pressure	220/8.9 ^{o}C/MPa

Reactor coolant system

Primary coolant flow rate	17.7 t/s
Reactor operating pressure	15.7 MPa
Coolant inlet temperature, at RPV inlet	289 ^{o}C
Coolant outlet temperature, at RPV outlet	321 ^{o}C
Mean temperature rise across core	32 ^{o}C

Reactor core

Active core height	3.53 m
Equivalent core diameter	3.16 m
Heat transfer surface in the core	5115 m^2
Fuel weight	70.1 t U
Average linear heat rate	16.7 kW/m
Average fuel power density	42.8 kW/kg U
Average core power density (volumetric)	109 kW/l
Thermal heat flux, F_q	586 kW/m^2
Enthalpy rise, F_H	
Fuel material	Sintered UO_2
Fuel assembly total length	3837 mm
Rod array	triangle
Number of fuel assemblies	163
Number of fuel rods/assembly	311
Number of guide tubes for control rods/instr.	18/1
Enrichment (range) of first core	1.6/ 2.4/ 4.0 Wt%
Enrichment of reload fuel at equilibrium core	4.0 Wt%
Operating cycle length (fuel cycle length)	12 months
Average discharge burnup of fuel	43 000 MWd/t
Cladding tube material	zirconium alloy
Cladding tube wall thickness	0.67 mm
Outer diameter of fuel rods	9.1 mm
Overall weight of assembly	680 kg
Active length of fuel rods	3530 mm
Burnable absorber, strategy/material	Partial compensation of reactivity reserve/ B_4C, zirconium matrix
Number of control rod assemblies	121
Absorber rods per control assembly	18
Absorber material	B_4C
Drive mechanism	Magnetic jack
Positioning rate	20 mm/s
Soluble neutron absorber	boron

Reactor pressure vessel

Cylindrical shell inner diameter		4136 mm
Wall thickness of cylindrical shell		200 mm
Total height		10885 mm
Base material:	cylindrical shell	15Kh2NMFA
	liner	stainless steel
Design pressure/temperature		17.6/350 MPa/^{o}C
Transport weight		320 t

<u>Steam generators</u>

Type	horizontal
Number	4
Heat transfer surface	6100 m^2
Number of heat exchanger tubes	10798
Tube dimensions	16x1.5 mm
Maximum outer diameter	4290 mm
Transport weight	235 t
Shell and tube material	10GN2MFA/0Kh18N10T

<u>Reactor coolant pump</u>

Type	GCNA-1391
Number	4
Design pressure/temperature	17.6/350 MPa/^{o}C
Design flow rate (at operating conditions)	5.97 m^3/s
Pump head	62±2,5 m
Power demand at coupling, cold/hot	6700/5000 kW
Pump casing material	Stainless steel
Pump speed	1000 rpm

<u>Pressuriser</u>

Total volume	79 m^3
Steam volume: full power/zero power	24/52 m^3
Design pressure/temperature	17.6/350 MPa/^{o}C
Heating power of the heater rods	2520 kW
Number of heater rods	28
Inner diameter	3000 mm
Total height	13600 mm
Material	10GN2MFA
Transport weight	197 t

<u>Pressuriser relief tank</u>

Total volume	30 m^3
Design pressure/temperature	0,6/150 MPa/^{o}C
Inner diameter (vessel)	2000 mm
Total height	7440 mm
Material	0Kh18N10T
Transport weight	17,8 t

<u>Primary containment</u>

Type	Dry, double wall
Overall form (spherical/cyl.)	cylindrical, in steel/reinforced concrete
Dimensions (diameter/height)	44/61,6 m
Free volume	68000 m^3
Design pressure/temperature (DBEs)	400/150 kPa/^{o}C
(severe accident situations)	600/200 kPa/^{o}C
Design leakage rate, inner shell	0.3 vol%/24 h

<u>Reactor auxiliary systems</u>

Reactor water cleanup,	capacity	8-16 kg/s
	filter type	ion exchange
Residual heat removal,	at high pressure	Removal of residual heat releases is effected through steam generators(SG) by steam release into turbine condensers and SG feed-up with flow 100-150 m^3/h
	at low pressure	800 m^3/h
Coolant injection,	at high pressure	up to 230 m^3/h
	at low pressure	800 m^3/h

<u>Power supply systems</u>

Main transformer,	rated voltage	24/500 kV
	rated capacity	2x630 MVA
Plant transformers,	rated voltage	24/6.3- 6.3 kV

	rated capacity	2x63 MVA
Start-up transformer	rated voltage	220/6.3- 6.3 kV
	rated capacity	2x63 MVA
Medium voltage busbars		6 kV
Number of low voltage busbar systems		28
Standby diesel generating units: number		4
	rated power	6.3 MW
Number of diesel-backed busbar systems		4
Voltage level of these		6 V ac
Number of DC distributions		18
Voltage level of these		110;220 V dc
Number of battery-backed busbar systems		16x220; 2x110.
Voltage level of these		110;220 V ac

Turbine plant

Number of turbines per reactor	1
Type of turbine(s)	K 100-6.1/50
Number of turbine sections per unit (e.g. HP/LP/LP)	1/3
Turbine speed	3000 rpm
Overall length of turbine unit	48,5 m
Overall width of turbine unit	16,5 m
HP inlet pressure/temperature	6,0/274,3 MPa/°C

Generator

Type	TZV-1100-2UZ
Rated power	1222,2 MVA
Active power	1068 MW
Voltage	24 kV
Frequency	50 Hz
Total generator mass	560 t
Overall length of generator	15.36 m

Condenser

Number of tubes	68000
Heat transfer area	65470 m^2
Cooling water flow rate	50 t/s
Cooling water temperature	20 °C
Condenser pressure	5.1 kPa

Condensate pumps

Number	3
Flow rate	600 kg/s
Pump head	2.7 MPa
Temperature	33 °C
Pump speed	1500 rpm

Condensate cleanup system

Full flow/part flow	-/30%
Filter type	mixed function

Feedwater tank

Volume	410 m^3
Pressure/temperature	1,13/185 MPa/°C

Feedwater pumps

Number	2 with turbodrive
Flow rate	982 kg/s
Pump head	775 m
Feedwater temperature	190 °C
Pump speed	4900 rpm

Condensate and feedwater heaters

Number of heating stages	4LPH+D+HPH

4.13.9 Measures to enhance economy and maintainability

The Atomenergoproject Institute has performed the comparative calculations of basic economical parameters for WWER-1000/V-392 and existing WWER-1000/V-320 plants. It was obtained that V-392 plant has the economical parameters considerably better than those of V-320 plant. In particular, specific capital investments in the construction of Novovoronezh NPP-2 are 1.6 times less than for a NPP of V-320 design. The electricity generation costs for Novovoronezh NPP-2 is about 1.5 times less than for existing serial plants with WWER-1000/V-320 design. At the same time the core damage frequency for Novovoronezh NPP-2 is about two orders of magnitude less than for V-320 plants (i.e., the WWER-1000/V-392 design ensures much higher level of the investments protection).

The improvement of the WWER-1000/V-392 economics has been mainly achieved due to combination of safety and normal operation functions. Thus, the emergency core cooling system performs both the safety and normal operation functions (primary coolant inventory maintenance at high and low pressure, containment spray, heat removal from the spent fuel pool). This design decision allowed to exclude four channels of the core HP emergency cooling system, four channels of containment spray system as well as the heat removal systems from spent fuel pool.

Use of the emergency heat removal system in the secondary circuit with the purpose to purify the secondary coolant allowed to exclude the associated normal operation systems which are used in existing NPP with WWER-1000/V-320 reactor plant. It should be noted that the use of safety systems with the purpose of normal operation results in decrease of operation cost for periodic inspection because these periodic inspections are made through the recurrent change of operating channels of such systems.

The second main factor for cost decrease is the extended use of passive systems that results in amount reduction of cables, equipment and components in the control and monitoring systems, the power supply systems, ventilation and service water systems.

Economical efficiency of NPP operation is determined by quantity of the produced power which depends on reliability of power production. The main index of reliability is the factor of power consumption that is the ratio of the quantity of power produced in a period of time (for instance during one year) to a quantity of power which could be produced if the unit would have rated power operation. Loss of power production can be a result of unit stop with the purpose of repair and maintenance including the change of spent fuel or because of occasional failures resulting in plant outage or power reduction. The plant availability goal for WWER-1000/V-392 design was established as 0.85 around the year. The factor of power generation losses due to repair and maintenance must not exceed 0.1 (that is the duration of the scheduled outage must not exceed 15–37 days per year). The factor of the power generation losses due to system failures must not exceed 0.05 per year. These goals have been met so that the estimated plant availability for Novovoronezh-2 is equal to 0,88.

4.13.10 Project status and planned schedule

The main design solutions of NPP-92 were used in the design of Novovoronezh NPP-2 in Russia and Kudankulam NPP in India which are presently under construction.

References

[1] General safety regulations for nuclear power plants (OPB-88/97).
[2] Nuclear safety rules for reactors of nuclear power plants (PBYA RU AS-89).

4.14 WWER-1000/V-466 REACTOR PLANT (GIDROPRESS, RUSSIAN FEDERATION)

4.14.1 Introduction

In developing the advanced WWER-1000/V-466 reactor plant (RP), the design of AES-91/99 was accepted as a basis. Development of the concept started in 1977 in co-operation between the specialists of the Saint-Petersburg Institute “Atomenergoproject” and the Finnish company Imatran Voima International Ltd (presently - Fortum Engineering Ltd).

The design is based on an evolutionary path of safety enhancement and at the same time it differs considerably from the prototype of the standard NPP with RP WWER-1000/V-320. In particular, there are some considerable differences in layout and configuration of the equipment, approaches to ensuring nuclear and radiation safety, and some main equipment has been modified.

Since 1991 when Russian organizations with design WWER-91 took part in a tender for the construction of the 5-th power unit in Finland for the NPP “Loviisa” site, work on improving the concept AES-91 has made an impressive headway. A series of meaningful design improvements has been introduced into the design of equipment and systems (reactor, steam generator, reactor coolant pump (RCP), reactor scram system, etc.) which have enabled to achieve the safety level meeting up-to-date requirements.

In 1991, the AES-91 design passed through the international examination in Saint-Petersburg where it was acknowledged as the design most ready of all for realization.

The design is based on safety criteria covered in normative-technical documentation valid in Russia as well as in IAEA recommendations.

The principle of defence-in-depth is the basis of safety assurance accepted in the design. It means application of the system of barriers on the way of propagation of ionizing radiation and radioactive substances into the environment and the system of technical and organizational measures for protecting barriers and maintaining their effectiveness, and directly for protecting the population.

Measures for protecting barriers involve:

- Use of up-to-date norms and standards;
- Consideration of operational feedback of standard reactor plant;
- Quality assurance at all the stages of RP life cycle;
- Monitoring for the state of barriers during operation by up-to-date means and methods of inspection;
- Maximum use of properties of reactor inherent safety peculiar to reactors of the WWER type;
- Control of technological process in such a way as to provide non-exceeding of design safety limits under all design conditions including accident conditions.

The design uses a deterministic approach assuming that safety is ensured in normal operation and at any initiating event considered in the design with regard to the single failure principle. Besides, the design provides measures on management of beyond design basis accidents and mitigation of their consequences.

In designing, a strategy was developed which is directed to prevention and, if it is not managed, to mitigation of severe accidents. In this case, a special reference is made to ensuring tightness of the containment because, in case of considerable damage to the containment, a large release of radioactivity into the environment cannot be be prevented.

The probabilistic analysis shows that design characteristics of the plant together with management measures of severe accidents provide the safety indices required by regulatory documentation. In particular:

- Frequency of severe core damage less than $1*10^{-5}$ per reactor-year;
- Frequency of limiting radioactive release less than $1*10^{-7}$ per reactor-year.

The design concept of AES-91 with WWER-1000/V-428 has been chosen for realization in constructing a NPP in China. By the present moment, DPR and PSAR have been developed, construction of power units of Tianwan NPP, Units 1&2 is under way. In this case, within the period from 1995 to 2001 numerous IAEA examinations were performed on the design documents of NPP with WWER-1000/V-428 for China.

The above mentioned reactor plant has an enhanced safety and improved feasibility indices that is provided by:

- Improvement of nuclear-and-physical properties of the core and structure of critical reactor units (transition to uranium-gadolinium fuel is assumed);
- Provision of reactivity coefficients to achieve negative feedback between power and coolant parameters;
- Introduction of new systems of monitoring and diagnostics of the equipment, primary pipelines and the valves;
- Improvement of RP special systems;
- Integration of the device for localization of core corium;
- Possibility of primary circuit depressurization in beyond design basis accidents as a result of integration of an additional line for PRZ PORV control from both main and emergency control rooms (feed and bleed);
- Use of removable thermal insulation of the modular type;
- Realization of the “leak before break” concept.

Further improvement of RP design for AES-91/99 is performed with regard to up-to-date requirements based on the requirements of Russian regulatory documents, requirements of European utilities (EUR), the Finnish safety authority (STUK), international standards and IAEA recommendations and it is developed according to the following basic trends:

- Reaching of service life of 60 years of the main equipment;
- Provision of maximum average burnup value of fuel assembly (FA) of 55 MW.day/kgU;
- Decrease in duration of outage and increase in load factor;
- Load follow and ability of manoeuvring;
- Possibility to use MOX-fuel.

4.14.2 Description of the nuclear systems

4.14.2.1 Primary circuit and its main characteristics

The reactor coolant circuit involves the reactor, four circulation loops, each of them consisting of a steam generator, a reactor coolant pump set as well as a main coolant pipeline connecting loops with the reactor. Basic characteristics of the reactor plant in steady-state modes under normal operation conditions are given in Table 4.14-I. A flow diagram of the reactor plant is represented in Figure 4.14-1. The principal layout of the reactor plant is shown in Figure 4.14-2.

4.14.2.2 Reactor core and fuel design

The reactor core is designed for generation of heat and its transfer from the fuel rod to the coolant during design service life without exceeding permissible fuel rod failure limits. The core comprises 163 fuel assemblies with uranium-gadolinium fuel which house absorbing rods of the control and protection system (CPS AR). The FA drawing is shown in Figure 4.14-3. The FA, as a part of the core, is intended for generation of thermal energy and its transfer to the coolant during design service life without exceeding permissible fuel rod failure limits.

TABLE 4.14-I. BASIC CHARACTERISTICS OF THE REACTOR PLANT

Technical characteristics	Value
1 Reactor thermal power, nominal, MW	3000
2 Number of loops, primary coolant system	4
3 Coolant flow rate through the reactor, nominal, m^3/h	86000
4 Coolant pressure at the reactor outlet, nominal, MPa	15,7
5 Coolant temperature at the reactor outlet, nominal, °C	321
6 Coolant heat-up in the reactor, °C	30
7 Steam capacity, t/h	5880
8 Generated steam pressure at the steam generator outlet, at full power, MPa	6,27
9 Primary system design parameters (for strength analysis)	
• pressure, MPa	17,6
• temperature, °C	350

CPS AR are intended for quick termination of the nuclear reaction in the core, power maintaining at the assigned level and its transition from one level to the other, core axial power field equalizing, prevention and suppression of xenon oscillations. The reactor scram system uses control rods (up to 121).

The standard WWER-1000 FA design has been improved to V-428 fuel assembly (FA-2) with the aim of creation of FA with increased stability to changing the shape during operation. The design of FA-2 has been developed with a rigid welded frame wherein the spacing grids are welded to zirconium guiding channels and the central tube. Work related to justification of the design of FA-2 of trial operation at operating reactors with WWER-1000 is under way.

The pitch electromagnetic drive ShEM-3 with position indicator is used for motion of control rods. Clusters of control rods contain 18 absorbing rods. Absorbing rods are made from stainless steel tubes filled with absorbing material that varies in length: boron carbide is used for the upper part and dysprosium titanate – for the lower part. Rod clad is sealed with face plugs. Control rod drives are mounted onto the reactor upper head. The drawing of ShEM-3 drive is represented in Figure 4.14-4.

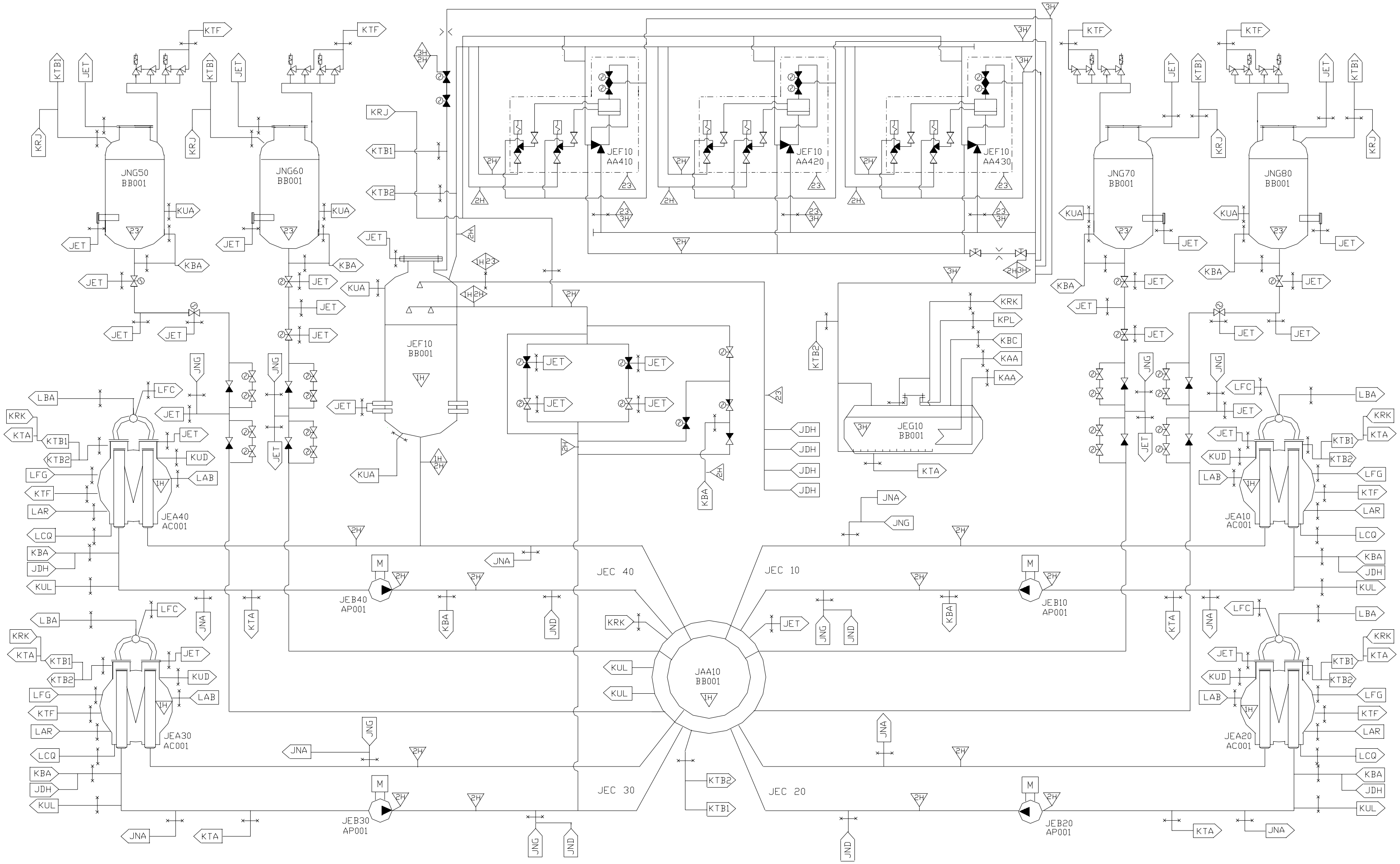


FIG. 4.14-1. Flow diagram of the reactor plant.

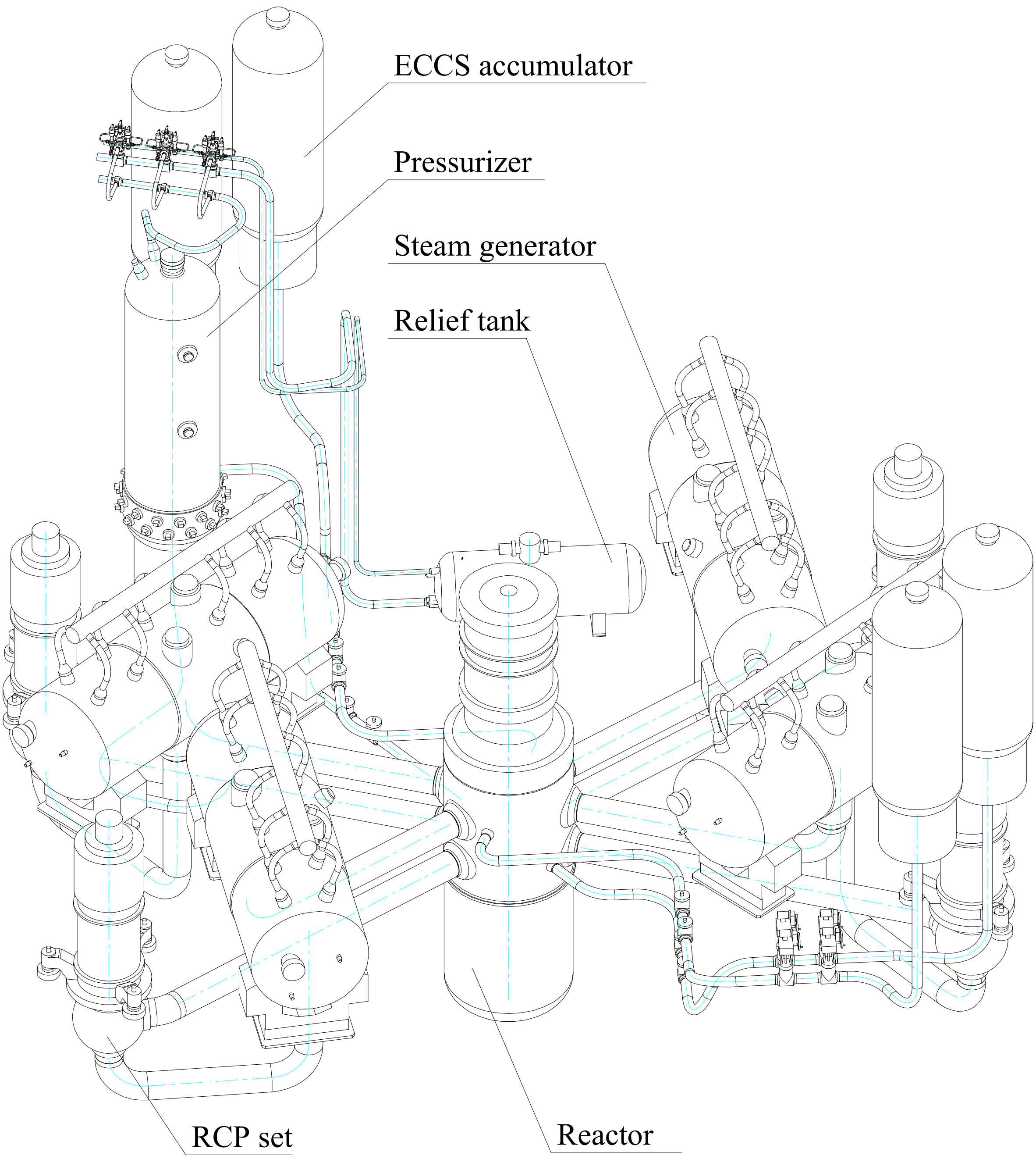


FIG. 4.14-2. Principal layout of the reactor plant.

TABLE 4.14-II. MAIN TECHNICAL CHARACTERISTICS AND OPERATION CONDITIONS OF THE CORE

Parameter	Value
Design nominal core power, MW	3000
Core pressure differential during operation of 4 RCP under nominal conditions, MPa	0,147
Average coolant velocity in the core, m/s, at the inlet	5,6
Coolant temperature at power operation, ^{o}C, nominal at the reactor inlet at the reactor outlet	 291 321
Nominal residence time of fuel assemblies in the core (cycle), year	4
Maximum permissible residence time of fuel assemblies in the core, eff.hours	30000
Effective time of fuel assembly operation between refuellings, effective hours, not less than	~ 8400
CPS CR drop time under scram, s	1,2 – 4,0
CPS AR motion velocity in control mode, m/s, nominal	0,02
Number of FA in the core, pcs.	163
Number of CPS CR drives, pcs.	Up to 121
Number of FA with CPS AR, pcs.	85
Core flow area, m^2	4,14
Equivalent core diameter, m	3,16
Pitch between fuel assemblies, m	0,236
Height of heated part (in cold state), m	3,53

The following has been done as compared with the drive of the standard power unit:

The power electromagnetic system is optimized and the dynamic drive characteristic is improved under motion conditions;
The drive service life is extended up to 30 years whereas the prototype mechanical part has a service life of 20 years, the service life of electromagnets is 10 years and that of the position indicator 5 years;
The pulling force is increased (twice as much);
The CR location measurement is improved with indication of pitch position each 20 mm of extension shaft motion (in standard WWER-1000 reactor, the drive provides indication of the position of the extension shaft with 350 mm pitch).

Work on structural optimization directed to reliability improvement, simplification and convenient maintenance of the ShEM-3 drive is continuously under way.

The results of operational-life proofs of the ShEM drive with CPS CR showed that the mechanical state of the drive and the extension shaft after proofs is satisfactory. The state and characteristics of FA and CPS AR during the proofs have practically not changed.

Safe fuel handling is ensured both by FA design and organizing measures. With this end in view, requirements have been established for fuel handling equipment as regards limitation of FA travel velocities during manufacture, transportation and loading into the reactor as well as usage of tilters, technological and transport containers.

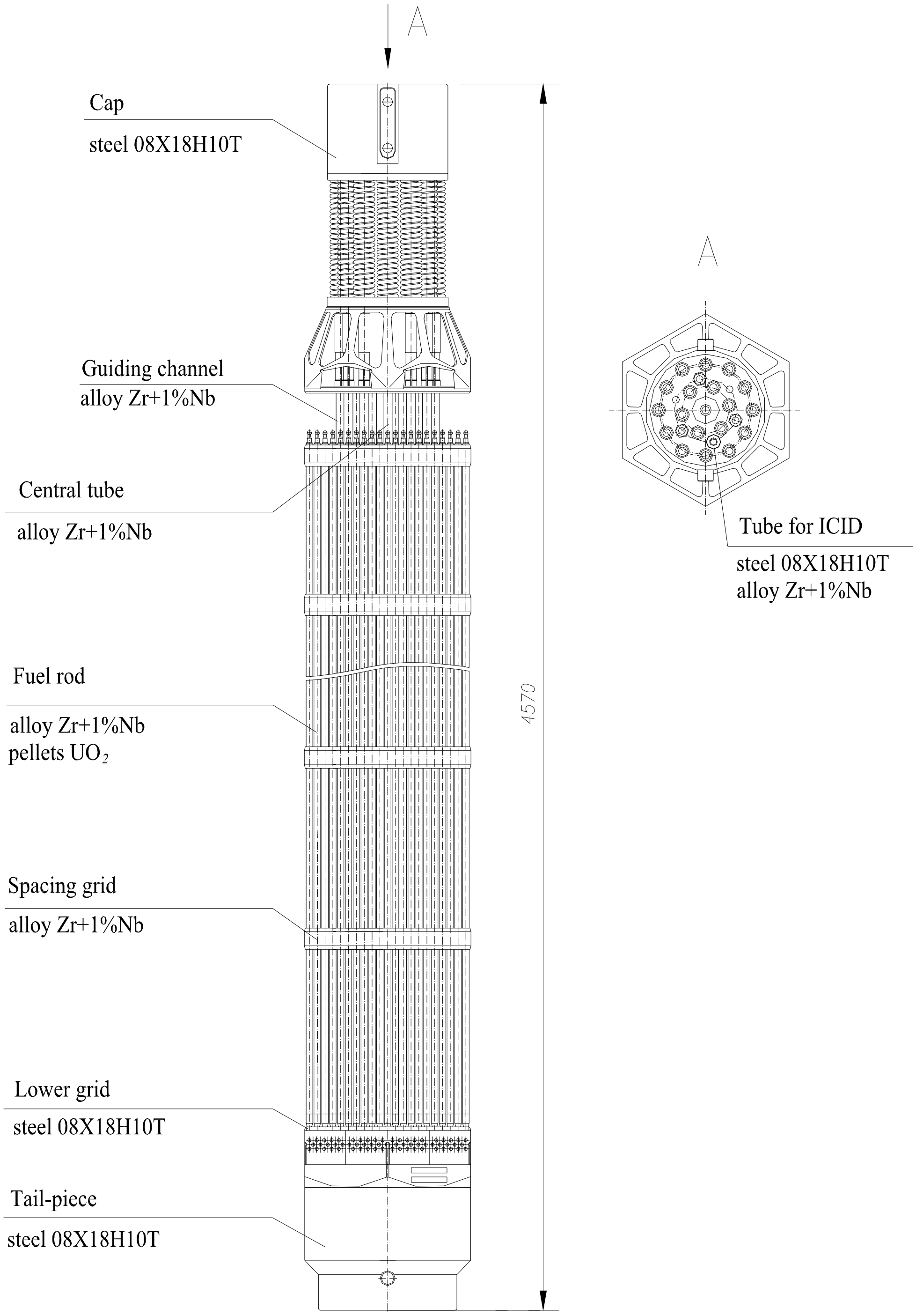


FIG. 4.14-3. Fuel assembly.

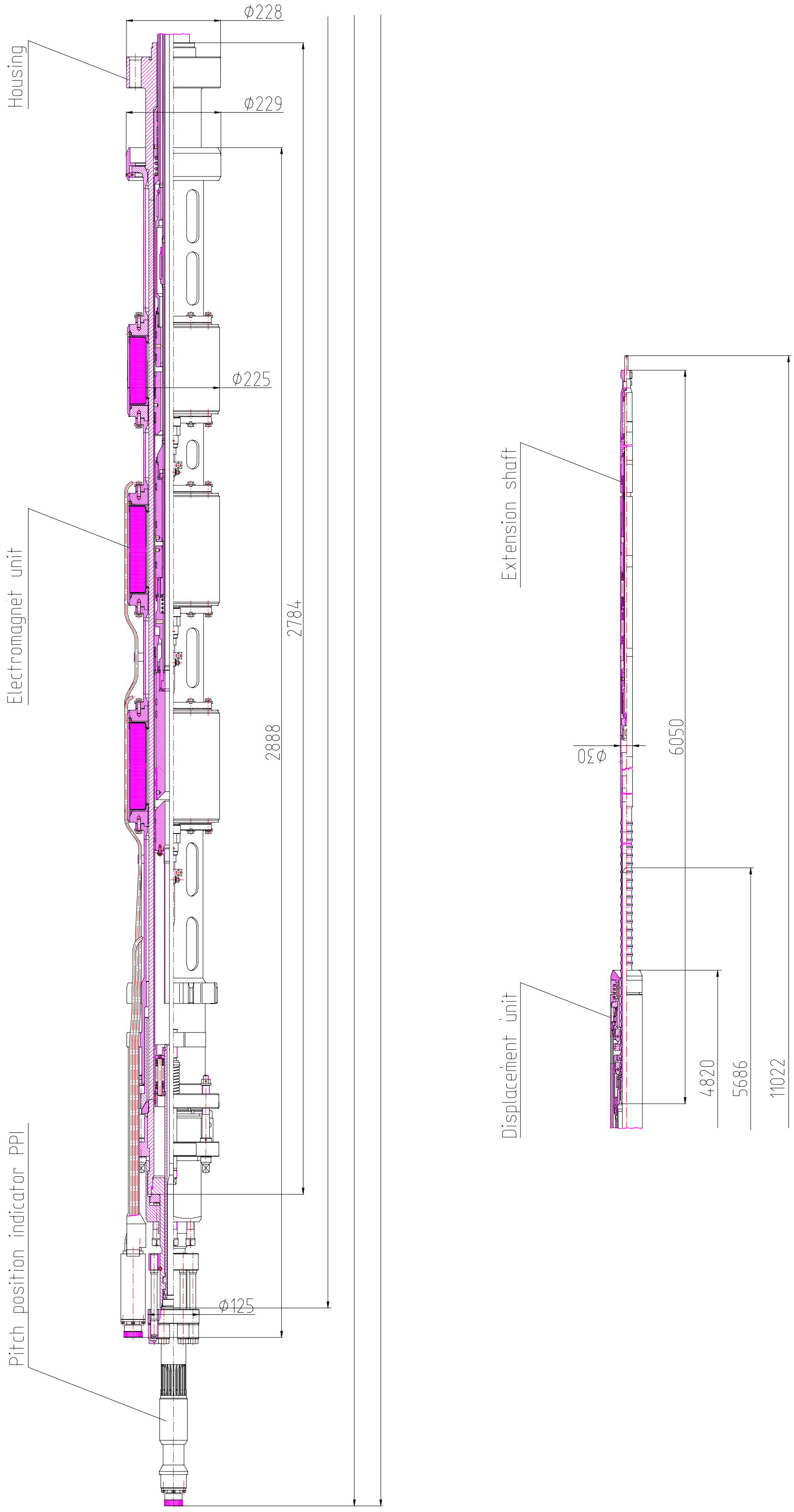


FIG. 4.14-4. CPS drive ShEM-3

4.14.2.3 Fuel handling and transfer systems

The fuel handling system involves systems of storage and handling with fresh and spent fuel and the reactor refuelling system.

The refuelling system is a complex of fuel handling equipment and it is intended for execution of procedures with fresh and spent fuel in the reactor compartment in the following sequence:

- Unloading of fresh fuel from the transport cask mounted in a multi-purpose seat;
- Unloading of spent FA (including clusters) into the racks of a spent fuel pond (SFP);
- Reshuffling of FA and clusters in the reactor core;
- Loading of fresh FA and clusters into the reactor core;
- Loading of spent fuel into the container for transportation from the reactor building.

Basic safety functions of the refuelling system are:

- Provision of nuclear safety (effective multiplication factor of neutrons shall not exceed 0,95);
- To prevent motion of cargoes above the stored fuel if they are not parts of hoisting and refuelling devices;
- Equipment stability at SSE and other natural phenomena;
- To prevent the FA from dropping and uncontrolled motion of mechanisms at safe shutdown earthquake (SSE) and in case of loss-of-power.

The following equipment are used for performing procedures during refuelling in the reactor building:

- fuel handling machine;
- transport cask;
- spent fuel pond racks;
- leak-tight bottle;
- multi-purpose seat;
- stem for covers of transport cask and container.

The fuel handling machine (FHM) is intended for the performance of procedures with FA, clusters and leak-tight bottles as well as for procedures with plugs of leak-tight bottles and bottles of the defective assembly detection system (DADS). Moreover, FHM makes possible to perform inspection of the axial position of FA caps in the reactor with the help of a special device and to perform examination of the mounting seat for FA.

The FHM consists of a bridge, a trolley, a working mast and a TV extension shaft. The FHM is made in seismic design version. In this case, the structure of the machine does not allow for its components and reloaded components to drop into the maintenance zone in case of an earthquake and loss-of-power.

The racks of the spent fuel pond (SFP) are intended for long-term storage of spent FA as well as for temporary storage of fresh FA. The SFP racks consist of separate sections. Each section for FA is a rigid rectangular stainless steel frame with the supporting perforated plate. Hexahedral borated tubes are mounted inside the frame. The rack section of leak-tight bottles is also made in the form of a rectangular stainless steel frame whose upper and middle part feature spacing plates with cells of a round form.

The leak-tight bottle is intended for a long-term storage of FA with leaky fuel rods. The bottle is a cylindrical welded structure comprising a case and a plug. The case is made in the form of a cylindrical shell with a bottom and a cap wherein cooling tubes are welded. From outside a bottle bottom features a catch, which holds the bottle mounted in the rack against turning during removal – mounting of the plug. The plug is intended for sealing the inside cavity of the case from the ambient medium. The plug has gripping fingers for transportation, sealing components and the valve intended for dropping pressure when working pressure is exceeded inside the bottle as well as for a short-term untightening of the inner cavity in removing and mounting the plug. The multi-purpose seat is a circular metalwork with a horizontal fitting surface and fixing slots. The multi-purpose seat is mounted in the SFP container compartment on the shock absorber and it is intended for mounting and orientation of the transportation cask and container for spent fuel in plan.

Refuelling procedures of the reactor are performed in the following sequence:

- Spent FA are withdrawn one by one from the reactor and installed into the cells of SFP racks by the fuel handling machine according to the refuelling cartogram;
- Reshuffling of FA and clusters is performed in the reactor;
- Then loading of fresh FA and clusters into the reactor is performed.

During the refuelling, inspection of tightness of fuel rod clads of FA unloaded from the reactor can be performed. In this case, FAs, in turn, one by one are mounted with the help of FHM into DADS bottles located in SFP for preparation and taking water sample. The FHM design provides for functioning of the leak check system of FA clads in FHM mast. It decreases the number of FA to be checked by the bottle method.

4.14.2.4 Primary components

Reactor

The reactor is a vertical pressurized vessel (vessel with top head) housing the internals, core, control rods and in-core instrumentation. The structure of the nuclear reactor assembled is shown in Figure 4.14-5. The reactor pressure vessel (RPV) is made of several solid-forged shells welded with each other and with an elliptical bottom and with the flange tightly sealed by two solid ring sealing gaskets reduced by 54 studs M170. Two nozzle shells of the vessel have four nozzles Dnom 850 each connected with the pipelines of the reactor coolant circuit. A ring separating the inlet chamber (in respect to coolant) from the outlet chamber, splines adjoining the core barrel and holding it from radial displacement and arms for mounting tight containers with surveillance-specimens of vessel type steel are welded on the inside vessel surface clad with austenitic coating.

The V-466 reactor pressure vessel is designed to ensure a life time of 60 years. The most important improvements are the location of surveillance specimens on RPV wall, reduction of the nickel content in welds, decrease of the content of harmful impurities in base metal and welds, decrease in $T_{ко}$ of nozzle shell (up to minus 35°C).

Modernization of the RPV is performed keeping design solutions for the upper part of the vessel (flange and nozzle zone). The geometry of the RPV bottom is also kept with the aim of using the well developed manufacturing process (i.e., reference character is practically maintained in respect to operating units of WWER-1000). At the same time, the inside diameter of the RPV cylindrical part is increased up to 4190 mm. It decreases the fast neutron flux to the RPV so that the total fluence during the operating period of 60 years does not exceed the neutron fluence for service life of 40 years of advanced V-428 and V-412 reactors.

Internals

The internals are intended for location of the core, control rod drives and in-core instrumentation as well as for arrangement of the coolant flow cooling fuel assemblies. The internals are a part of the reactor and involve the core barrel, core baffle and protective tube unit (PTU). The core barrel provides for keeping the core initial geometry by spacing each fuel assembly both in the fixed cells of the spacing grid of the core barrel and in the spacing grid of the protective tube unit. The core barrel receives loads from FA, the core baffle and protective tube unit and it receives thermohydraulic forces from moving coolant. The PTU provides arrangement of control rods and in-core instrumentation detectors as well as their protection from the dynamic effect of the coolant flow. The core baffle decreases the value of neutron flux to the reactor vessel and is “a displacer” between the core barrel and peripheral FA.

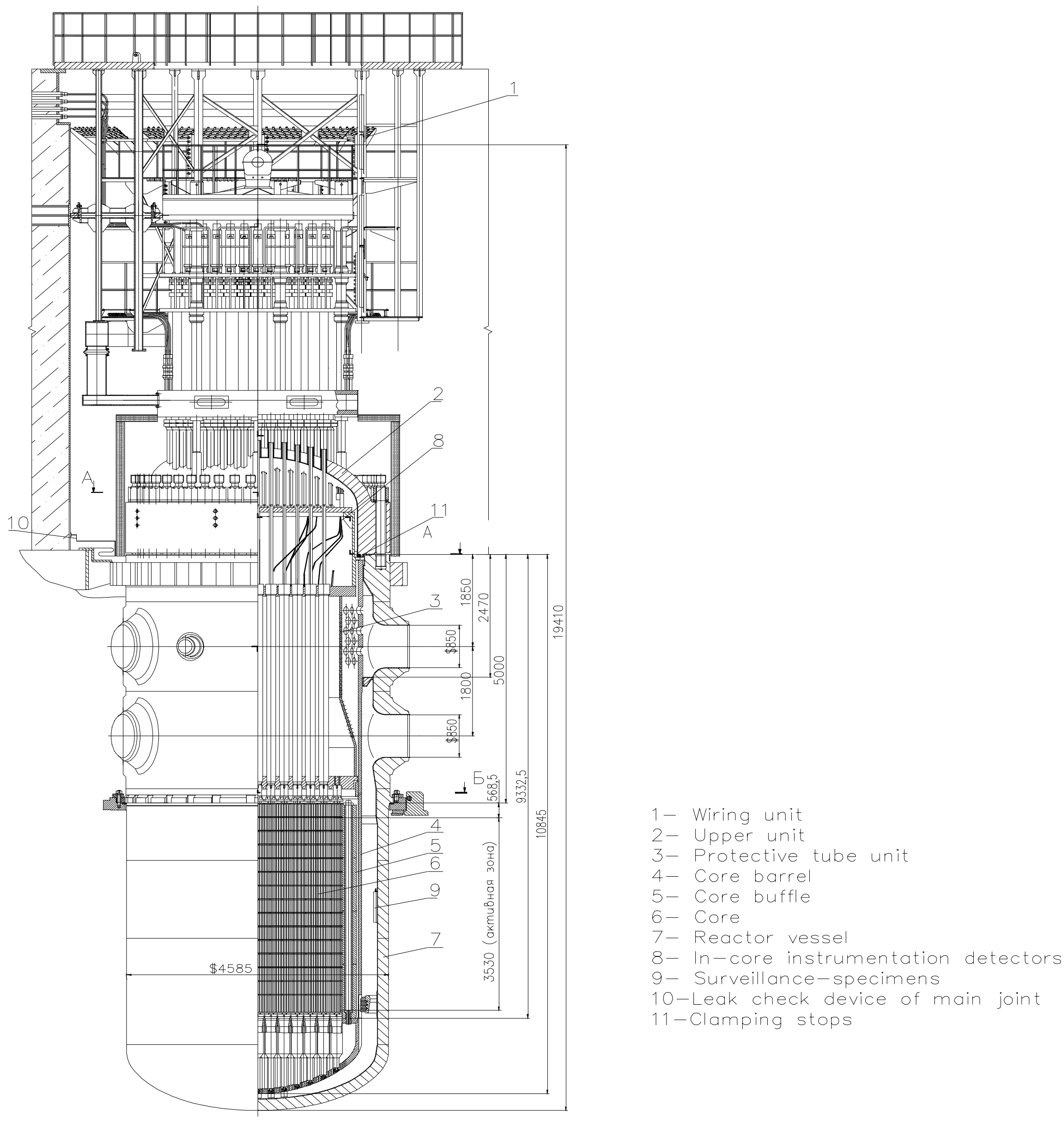


FIG. 4.14-5. Reactor assembled.

The perforation of the core barrel and PTU shell decreases pulsation and non-uniformity of the coolant flow rate at the core inlet and outlet as well as it improves coolant mixing (by boron concentration and temperature) supplied into the reactor from different loops by operating reactor coolant pumps.

Structural joints of the reactor and internals components restrict displacement of FA and internals under the coolant flow load and pressure differential under all design basis conditions. Free thermal displacements at power operation and withdrawal from the vessel in case of refuelling and examinations is provided.

Steam generator

The steam generator (see Figure 4.14-6) is of a horizontal type - traditional for WWERs - with a submerged heat exchanging surface.

The problem of ensuring the reliable operation of PGV-1000 tubing isrelated, primarily, to the state of water chemistry of the WWER-1000 power units. Under violation of water chemistry as well as under undue chemical cleaning there are cases with damages of heat exchanging tubes related to their increased fouling with operational deposits and clogging of separate zones of intertube space with sludge.

Practice has shown that the steam generators of the WWER-440 type with corridor arrangement of the tube bundles have a smaller number of heat exchanging tubes plugged than the steam generators of the WWER-1000 type with a staggered arrangement of tubes in the bundle. It should be noted that the operating life of steam generators of the WWER-440 type is twice as much. Also, at the initial period PGV-440 operated at rather worse quality of feed and blowdown water than PGV-1000 did.

EDO "Gidropress" has developed the PGV-1000M steam generator with an increased diameter of the vessel and sparse corridor arrangement of tubes in the heat exchanging bundle. This design feature is intended to improve the operational reliability of heat exchanging tubes and to maintain the water chemistry of SG secondary circuit in conformity with the requirements accepted by leading nuclear countries.

Horizontal steam generators used at NPP with WWER have a series of advantages as compared with SG of other types. The following specific features stipulating their reliability and safety as a part of RP should be noted:

- Moderate rate of evaporation to steam relieving area (0,2–0,3 m/s) which makes possible use of a simple separation scheme under the reliable provision of the required steam moisture content;
- Moderate velocities of the medium in the secondary side (up to 0,5 m/s) practically exclude the cases of tube damage from mechanical effect of foreign objects which may happen to be in the cavity of the secondary side. Studies of vibration of heat exchanging tubes have shown that the secondary coolant flow introduces insignificant contribution into the vibration level which is small even without it;
- With the assumed water chemistry, use of stainless austenitic steel 08Kh18N10T with low nickel content is justified. Its serviceability is confirmed by operational feedback of steam generators WWER PGV-440 and PGV-1000 (operating life is over 30 years);
- The ratio of wall thickness to the diameter of heat exchanging tubes is considerably greater than in steam generators of other designs which is a positive factor in terms of load-bearing capability of the tube wall. There was not a single case of break of heat exchanging tubes during the whole period of operation of horizontal SGs;
- The vertical primary-side cylindrical collector enables to avoid accumulation of sludge deposits on them and as a result of this decrease in corrosion damage of the coils in the area of their attachment places into the primary-side collector. In PWR vertical SGs, the heat exchanging tubes are secured into the horizontal flat tube plate and they have an increased probability of damage of the whole bundle in the place of their attachment to the tube plate;
- The steam generator has an increased water inventory in the secondary side as compared with known operating steam generators of the vertical type. It favours a more reliable cooldown of the reactor through SG in case of cessation of normal and emergency water supply. The high SG accumulating capability also mitigates reactor plant transients;
- An important advantage of horizontal SG is the possibility of application of the principle of step-wise evaporation in it which helps to maintain the concentration of the dissolved impurities in SG crucial areas several times below than the balanced concentration in blowdown water that essentially raises the reliability of SG operation in terms of corrosion;
- The steam generator provides a reliable natural circulation in the primary circuit even if the secondary water level decreases below the upper rows of the heat exchanging tubes;
- The horizontal arrangement of the heat exchanging surface is favourable in terms of provision of the primary coolant natural circulation under accident conditions. In this case, accumulation of non-condensable gases is prevented. These gases hinder natural circulation. Availability of big gas-holders (up to 0,5 m^3) in the upper part of the coolant collector arranged vertically facilitates dumping of gas from the tube bundle into the gas removal system;
- Rather convenient access to tubing for maintenance and inspection both from the primary side and the secondary side is provided. It also facilitates removal of foreign objects from the

secondary side cavity. There are no heat exchanging tubes in the lower points of the SG vessel where deposition and accumulation of sludge is possible. In case of accumulation of corrosion active impurities in the vessel lower part with consequences that are dangerous for the tubes, their washing away is possible via the SG blowdown system;

- All welds of the SG units subject to pressure are accessible for inspection and repair foreseen in design documentation, and repair;
- A complex of inspection, diagnostic and repair equipment providing possibility of maintenance and preventive removal of detected defects.

Use of sparse corridor arrangement of tubes in heat exchanging bundle makes it possible to:

- Increase the circulation rate in the tube bundle that will reduce the probability of damage to heat exchanging tubes due to decrease in the growth rate of deposits on the heat exchanging tubes and the degree of concentration of corrosion-active impurities under them;
- Reduce the possibility of clogging the inter-tube space with spalled sludge;
- Provide for easy access into intertube space for inspection of heat exchanging tubes and their cleaning if necessary;
- Increase the water inventory in the steam generator;
- Enlarge the space under the tube bundle for easy removal of the sludge.

Pressurizer

The pressurizer (PRZ), as it is shown in Figure 4.14-8, is a vertical vessel mounted on the cylindrical support. The pressurizer consists of the vessel, internals, support, tubular electric heater units, surge tanks. The PRZ vessel involves a cylindrical shell, zone of heaters, two semi-ellipsoidal heads and support.

The top head furnishes a manhole for examination of the inside surface, two nozzles for injection pipelines, and a nozzle for pipeline of dumping through pilot-operated relief valves. The hatch nozzle has sleeves for tightness check, for surge tanks, and an air vent.

The bottom head includes a nozzle for the mounting pipeline connecting “hot” leg of the primary loop with PRZ as well as four sleeves for surge tanks.

The cylindrical part of the vessel has fourteen sleeves for level gauges and two sleeves for sampling.

The dimensions of the pressurizer and the steam-water volume ratio are chosen in such a way as:

- To prevent from coolant boiling in any point of the primary circuit, actuation of pilot-operated relief valves, uncovering of electric heaters under normal operation conditions and anticipated operational occurrences;
- To provide acceptable characteristics of pressure variation under variation of the system volume during load transients;
- To provide sufficient power of electric heaters for heating up PRZ, filled with water at zero power, at the rate providing temperature (and, respectively, pressure) in PRZ;
- To ensure that steam ingress into the primary circuit from PRZ shall not take place under any non-LOCA design conditions.

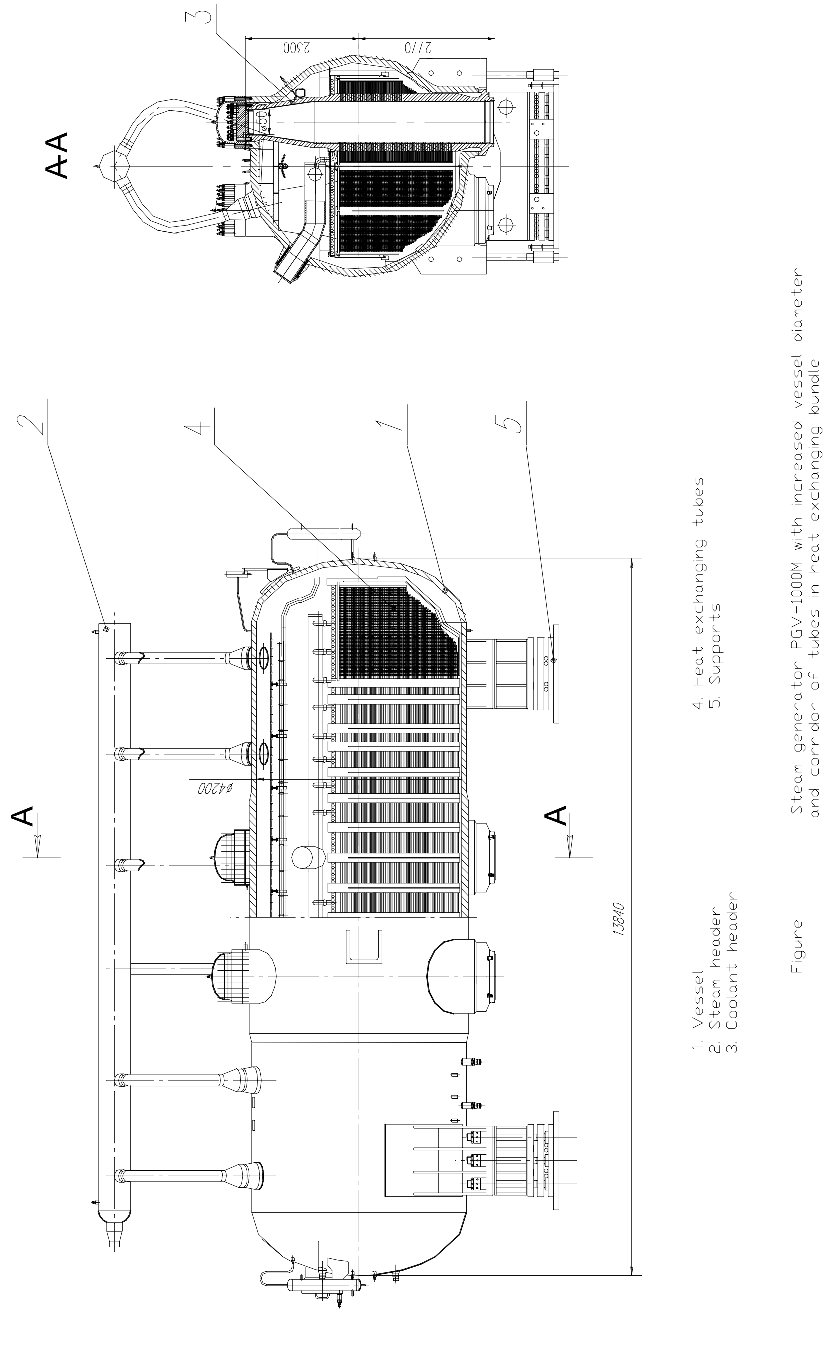


FIG. 4.14-7. Steam generator PGV-1000M with increased vessel diameter.

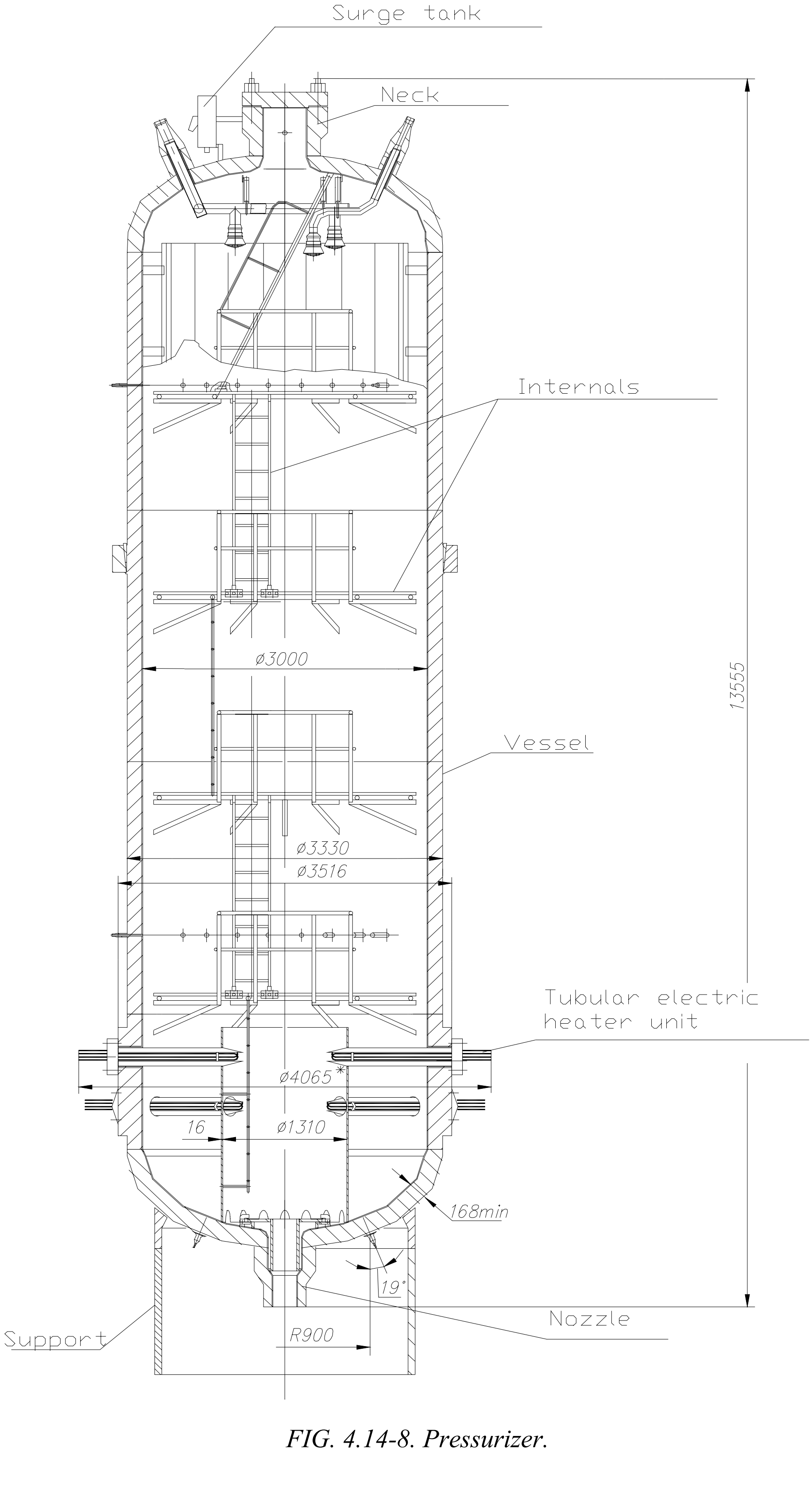


FIG. 4.14-8. Pressurizer.

The PRZ is designed so that rise of water level shall not take place up to the limits under which water is released through safety valves in design modes (with the exception of leaks from the PRZ upper part, spurious injection into the PRZ from the makeup system and transients with failure to scram).

The combination of thermal losses from the PRZ and the power of electric heaters shall maintain pressure in the PRZ at the nominal level under the conditions of a hot shutdown.

The PRZ together with the chemical and volume control system (KBA) provides for depressurization in the primary circuit during cooldown.

Reactor coolant pump set

The reactor coolant pump (RCP) represents a vertical centrifugal one-stage pump set GTsNA-1391 (see Figure 4.14-9) comprising a hydraulic case, removable part, electric motor, upper and lower spacers, supports and auxiliary systems.

The reactor coolant pump is intended for creating coolant circulation in the primary circuit of the reactor plant. The RCP has an additional function for providing coolant circulation during coastdown at various accidents with loss-of-power. This provides a reliable core cooling in case of breaks in power supply up to 2 seconds and it improves conditions of reaching natural circulation.

The design of the reactor coolant pump is developed proceeding from the condition of a reliable fulfilment of its functions and maintaining strength under impact of operational, seismic and accident loads. These loads and their combinations are taken into account with regard to the number of their cycles and service life covered in the design of the reactor plant. Besides, non-exceeding of loads from the side of RCP to its related systems is ensured by RCP design.

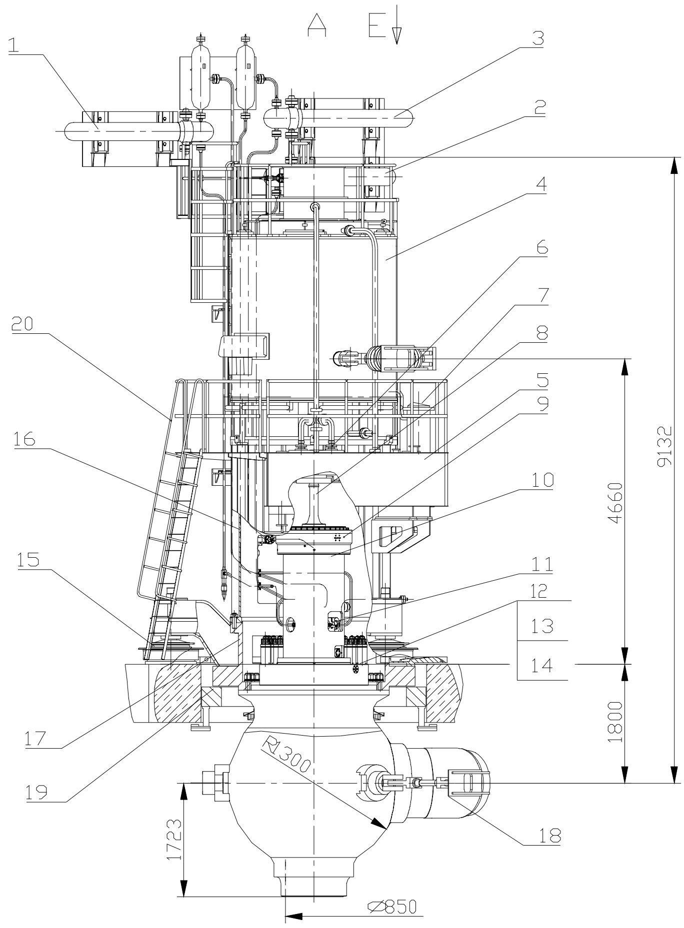
With the reactor coolant pump set GTsNA-1391, the following reliability requirements can be met:

- Average service life is not less than 40 years;
- Continuous operation of GTsNA-1391 under all operating conditions amounts to not less than 12000 hours;
- Average service life between medium repairs is not less than 24000 hours;
- GTsNA-1391 is maintainable, it offers the possibility for disassembling and replacing of the components, average failure interval is not less than 18000 hours.

In designing GTsNA-1391, the operational experience of GTsN-195M pump was widely used. The latter operates presently at all existing power units with WWER-1000 reactors.

The GTsNA-1391 as compared with the prototype has the following advantages:
- Use of water as lubricant and cooling of the main thrust bearing of the pump and incombustible lubricating liquid in electric motor as well as plate coupling not requiring lubricant that makes possible not to use the oil system and completely exclude fire hazard;
- Use of an independent cooling circuit of the lower bearing operating during RCP set outage by the natural circulation principle that makes possible not to use auxiliary pump;
- Use of RCP set with sealing the shaft that prevents from coolant leak during loss-of-power for 24 hours enables to keep tightness of the reactor coolant circuit with no supply of RCP sealing water.

Design changes performed in GTsNA-1391 as compared with GTsNA-195M are directed to the enhancement of reliability and independent operation with respect to the plant systems.



Poz	Name	Quan
1	EC Cooler	1
2	SU Cooler	1
3	RAB Cooler	1
4	Electric drive	1
5	Oil tank	1
6	Electric pump	2
7	Oil Cooler	1
8	Torsion coupling	1
9	Electromagnet	1
10	Radial-axial bearing	1
11	Sealing unit	1
12	Casing	1
13	Lower radial bearing	1
14	Impeller	1
15	Roller support	3
16	Upper spacer	1
17	Lower spacer	1
18	Spherical casing	1
19	Biological shielding ring	1
20	Service platform	1

FIG. 4.14-9. Reactor coolant pump set GTsNA-1391.

Main coolant pipeline

The main coolant pipeline (MCP) of the reactor plant is similar to the pipeline of the V-428 reactor plant applied to Tianwan NPP in China. The small difference in MCP geometry is pertained to the use of a steam generator with corridor arrangement of bundle with a diameter of 200 mm more than that of the standard SG.

For MCP, the “leak before break” (LBB) concept is implemented that implies the absence of emergency restraints on MCP and emergency tie-rods on the reactor coolant pump. Materials used for making pipe units, and electrodes applied for the welds, are analogous to those applied in the standard design and in the design of Tianwan NPP in China.

4.14.3 Description of the turbine generator plant systems

No information provided.

4.14.4 Instrumentation and control systems

No information provided.

4.14.5 Electrical systems

No information provided.

4.14.6 Safety concept

No information provided.

4.14.7 Plant layout

No information provided.

4.14.8 Technical data

See Section 4.14.2.

4.14.9 Measures to enhance economy and maintainability

The V-466 RP design is developed taking into account the latest requirements and achievements in nuclear power. On the one hand, it is to a great extent a reference design concerning the equipment and systems being used at the operating NPPs with WWER-1000 as well as at the NPP under construction in China. This permits to decrease the RP cost due to standardization and unification of the applied equipment, process components and materials as well as the experience of designing, manufacturing, construction and operation. The technical solutions made in the design provide the necessary level of reliability and safety of RP due to the balanced number of active and passive safety systems. These solutions ensure also the adequate prevention and mitigation of the design basis and severe accidents that is confirmed by IAEA expert examination of RP design for the NPP in China.

On the other hand, a number of new technical solutions were incorporated into the design, the results of which ensure an essential improvement of economic indices of the reactor plant. First of all, it is the increase in the service life of the main equipment (reactor and steam generator) that leads to a decrease in expenses for its manufacturing, transportation, storage, preservation, etc.

A number of solutions directed toward simplification of the design are implemented. For example, in the RCP set the usage of water as lubricant and cooling of the main thrust bearing of the pump, and use of a fire resistant lubricating fluid in the electric motor as well as a plate coupling that does not require lubrication permits not to use the external oil system. The usage of an independent circuit for cooling the lower bearing, which, in case of RCP set outage, operates by the principle of natural circulation, permits not to use the auxiliary pump, etc.

Besides, as a result of the analysis of operation experience of NPPs with WWERs, in the V-466 RP design, measures are implemented which are directed at decreasing the number of failures of RP components leading to an increase in the capacity factor. Thus, the corridor layout of tubes in the SG heat exchange bundle ensures the following:

- Improvement of circulation in the tube bundle compared to a standard PGV-1000 having a staggered arrangement of tubes;
- Decrease of a probability of choking the shell-side with separated sludge;
- Improvement of conditions for chemical and mechanical cleaning;
- Decrease in the degree of concentration of corrosive-active impurities in the deposits and as a result a decrease in the probability of damage to heat exchanging tubes.

The measures improving the reliability and decreasing the probability of failure of control rod drive due to modernization of electromagnet units and the position indicator are also implemented.

In the V-466 RP design, a reduction in maintenance and repair time is provided together with an increase in FA burnup. These factors, and the optimization of the fuel cycle and other measures lead to an increase in the plant capacity factor.

The decrease in maintenance and repair timeresults from:

- Reduction of the scope and manpower needed for these operations;
- Improvement of technologies for maintenance and repair;
- Improvement of the corresponding operating documentation;
- Use of the up to date equipment with improved characteristics of reliability (increased service life, improved reparability, etc);
- Improvement of logistics, repair quality and information support.

The reduction in duration of scheduled shutdowns of RP for refuelling, inspection, maintenance and repair results from:

- Optimization of the repair cycle structure, including increase in the period between repairs;
- Improvement of organization for preparation and performance of maintenance and repair, and improvement of personnel training quality;
- Optimization of the work scope, optimization of conducting operating conditions under RP shutdown;
- Application of high-efficiency means of technological equipment for maintenance and repair (automated monitoring systems, multi-position power-nut drivers, repair equipment, etc);
- Increase in the rate of performance of fuel handling procedures (increase in velocity of fuel handling machine and polar crane in the reactor hall);
- Improvement of maintenance and repair technologies.

As a whole the measures implemented in the V-466 reactor plant design together with the relevant plant measures permit to lower greatly the cost of the supplied electric power in relation to operating units with WWER-1000.

4.14.10 Project status and planned schedule

The WWER-1000/V-466 reactor plant design is developed in sufficient details to be applied and licensed in the frame of an advanced NPP.

4.15 WWER-1500/V-448 REACTOR PLANT (GIDROPRESS, RUSSIAN FEDERATION)

4.15.1 Introduction

In 1996, "Technical requirements for large power unit for wide-area implementation after the year 2010" were developed by the leading organizations of Ministry of Russian Federation for Atomic Energy (MAE). Based on the specified requirements OKB "Gidropress" has developed a conceptual design of the WWER-1500 reactor plant. The conceptual design makes use of the long-term experience gained in development and operation of the WWER-440 and WWER-1000 plants. The design has confirmed in principle the possibility for development of the 1500 MW(e) reactor plant on the existing process equipment of Russia industry with its minimum modernization, with significant improvement of safety level, reliability, as well as economic indices, and is recommended by MAE for further development.

At present, the detailed design is under development. Safety assurance of personnel, public and environment against radiation exposure is used as the design basis. The specified doses and source terms shall not be exceeded during normal operation, anticipated operational occurrences, design-basis accidents (DBA) and beyond design-basis accidents throughout the NPP service life (50 years).

The allowable levels for fuel rod damage and radioactivity release are specified in the national regulatory standards and in the Technical Assignment for WWER-1500/V-448 reactor plant.

The operational limit with regard to fuel rod damage is 0.2% of fuel rods with the gas leak-untightness and 0.02% of fuel rods with the direct fuel-coolant contact. Damage to the fuel rods above the specified values as regards radioactivity release out of the fuel rods and steam-zirconium reaction, being the maximum design limits, shall not be exceeded during the DBA.

The estimated probability for severe core damage, including that for the shutdown reactor, shall be less than 10^{-6} per reactor-year. The estimated probability for limiting release of radioactive products into the environment leading to necessary evacuation of the population from the area prescribed by the relevant guides is specified to be less than 10^{-7} per reactor-year.

The NPP safety shall be provided by permanent implementation of the "defense-in-depth" principle based on application of the system of barriers on the path of radioactive substances propagation into the environment as well as the systems of engineered safeguards and of organizational provisions for protection of these barriers. The permanent implementation of the "defense-in-depth" principle implies:

- Application of consecutive physical barriers on the path of propagation of radioactive substances: fuel matrix, fuel rod cladding, primary circuit, containment,
- Consideration of all postulated initiating events that could result in loss of efficiency of these barriers;
- Determination, for each postulated event, of design measures and actions of the operating personnel required to keep integrity of the above barriers, and to mitigate the consequences of damage to such barriers;
- Minimization of probability of the accidents resulting in radioactivity release; and
- Redundancy and diversity of the safety systems, and physical separation of the safety system trains.

The reactor plant considered is of the evolutionary type. The design is developed in accordance with the latest Russian safety regulations for NPP that meet the modem world requirements. The design organizations such as OKB "Gidropress", Russian Research Centre "Kurchatov Institute", AEP are the well-known designers of WWER plants. Quality assurance requirements of the IAEA and the international standard ISO 9000 are taken into account in the design as well.

The principal distinctions characterizing the safety philosophy accepted in the design are as follows:

- Significant reduction of specific power (as compared with V-1000) owing to increase of number of FA;
- The neutron fluence onto the reactor vessel for 60 years is three times less than the fluence onto the reactor vessel of the standard V-1000 for 40 years;
- Possible maintaining of core sub-criticality using the Core Protection System (CPS) at any moment of lifetime during coolant temperature decrease to 100 °C and assuming complete replacement of boric acid in the primary circuit with pure condensate;
- Application of passive system for residual heat removal from the reactor plant in case of NPP blackout;

4.15.2 Description of the nuclear systems

4.15.2.1 Primary circuit and its main characteristics

The reactor plant WWER-1500 is a constituent part of large power unit. The reactor plant (RP) represents itself the pressurized primary circuit and associated systems and equipment for its maintenance placed within the containment, and also a complex of the systems for monitoring, control, regulation, protection, interlocks, signaling and diagnostics intended to form the automatic process control system (APCS) for the reactor plant.

The leak-tight primary circuit is the next protective barrier after the fuel rod cladding used to restrict propagation of radioactive substances during the accidents.

The primary circuit is a circulation circuit under operating pressure together with the pressurization system.

The primary circuit together with the associated systems provides heat removal from the reactor core by the coolant during normal operation, anticipated operational occurrences and design-basis accidents, and heat transfer to the secondary circuit through the heat-exchanging surface in the steam generator. The circulation circuit has a traditional four-loop configuration with reactor, four horizontal steam generators, four RCP sets and connecting pipelines. The flow diagram of the reactor plant is given in Figure 4.15-1.

The primary equipment and pipelines and associated systems as well as the sections of pipelines and systems intended to localize the active coolant during the accidents are arranged within the inner protective shell (made of pre-stressed ferro-concrete) 49 m in diameter. The latter is designed to take up the internal impacts and, in turn, is protected by the outer enclosure (of non-stressed concrete) to take up the external impacts and to decrease the parameters of dynamic impacts upon the equipment arranged within the inner protective shell. Layout of the RP equipment within the containment is shown in Figures 4.15-2, 4.15-3. The main characteristics of the RP equipment are given in Section 4.15.8.

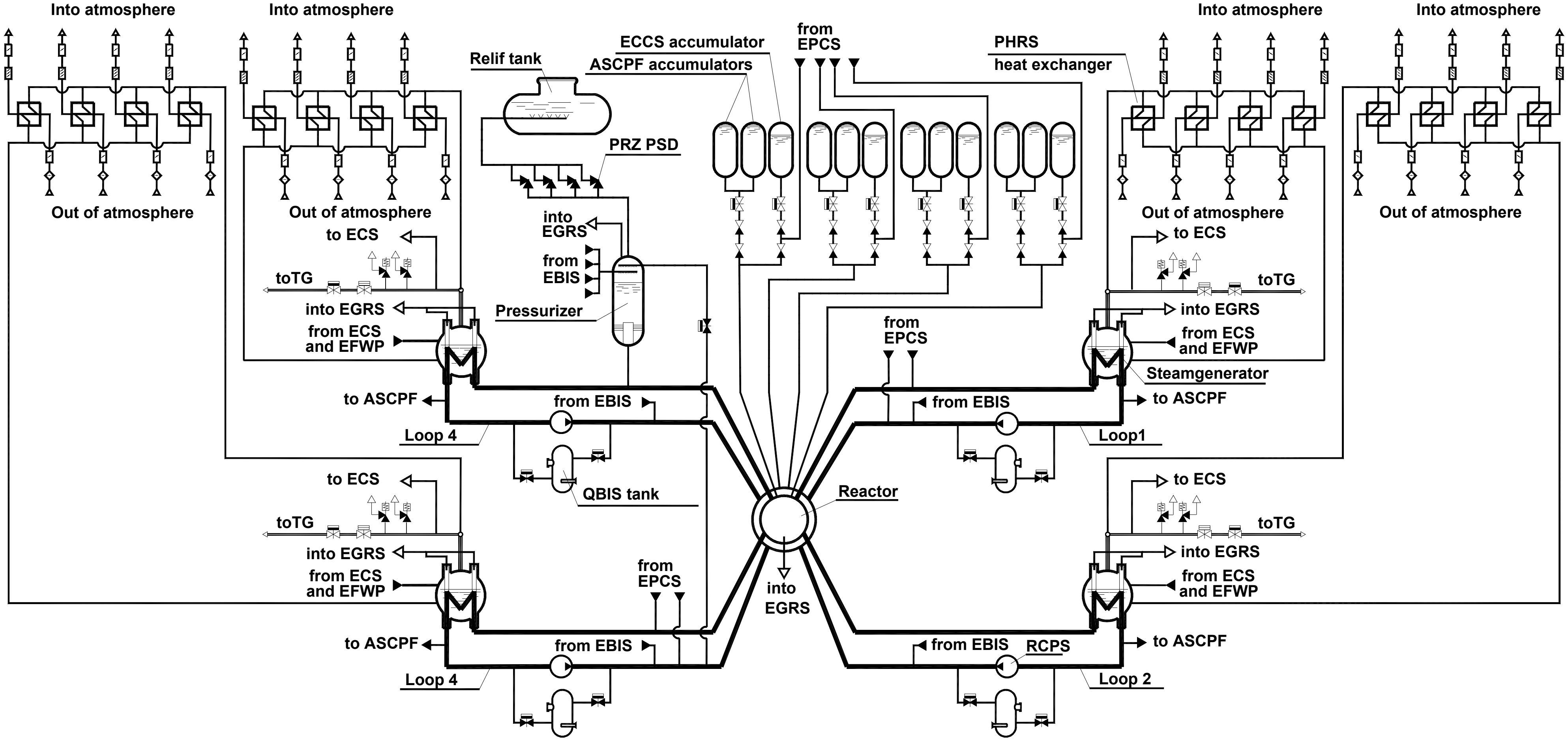


FIG. 4.15-1.   Flow diagram of the RP V-448 primary circuit

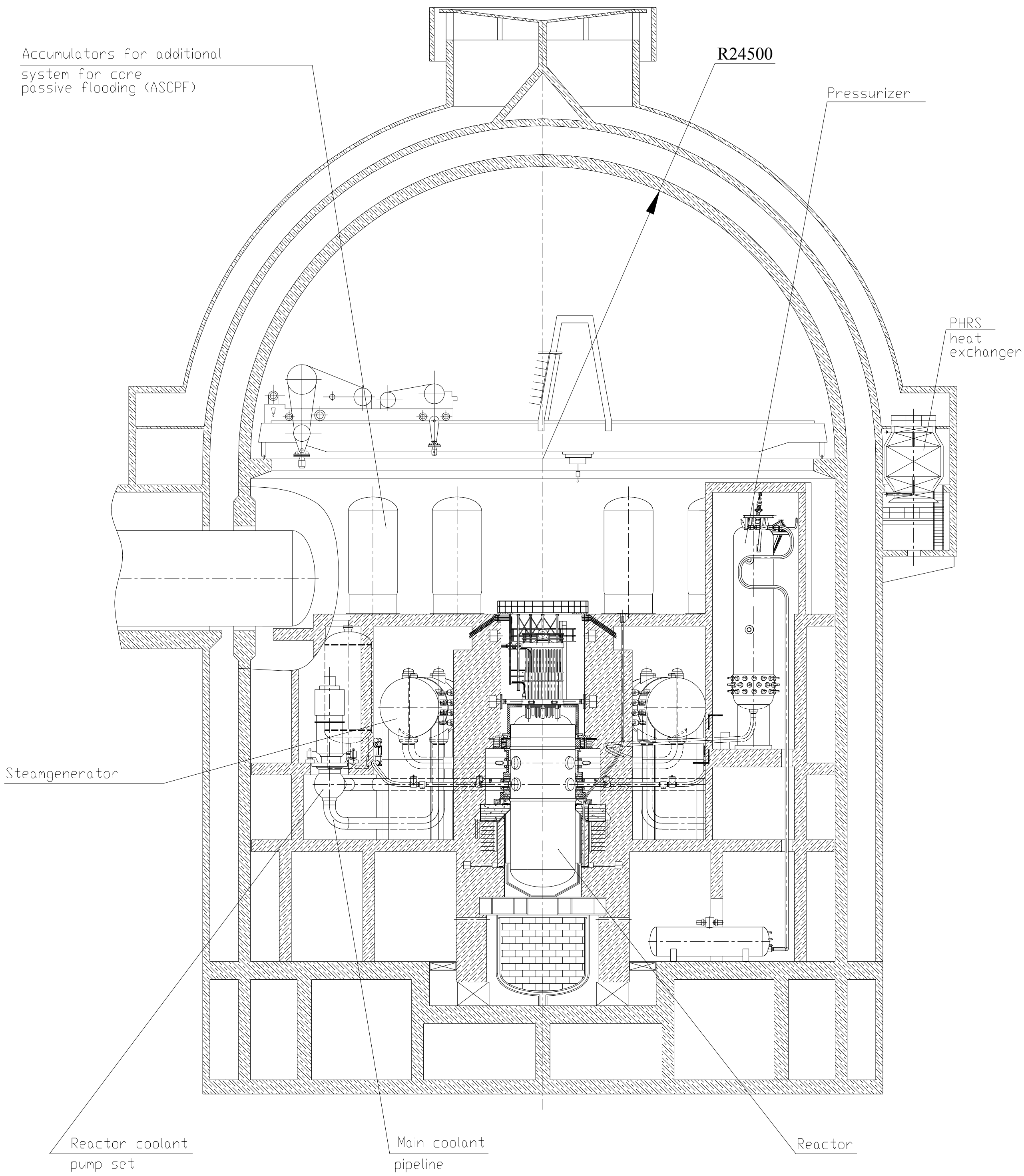


FIG. 4.15-2. Layout of the RP V-448 equipment (longitudinal section)

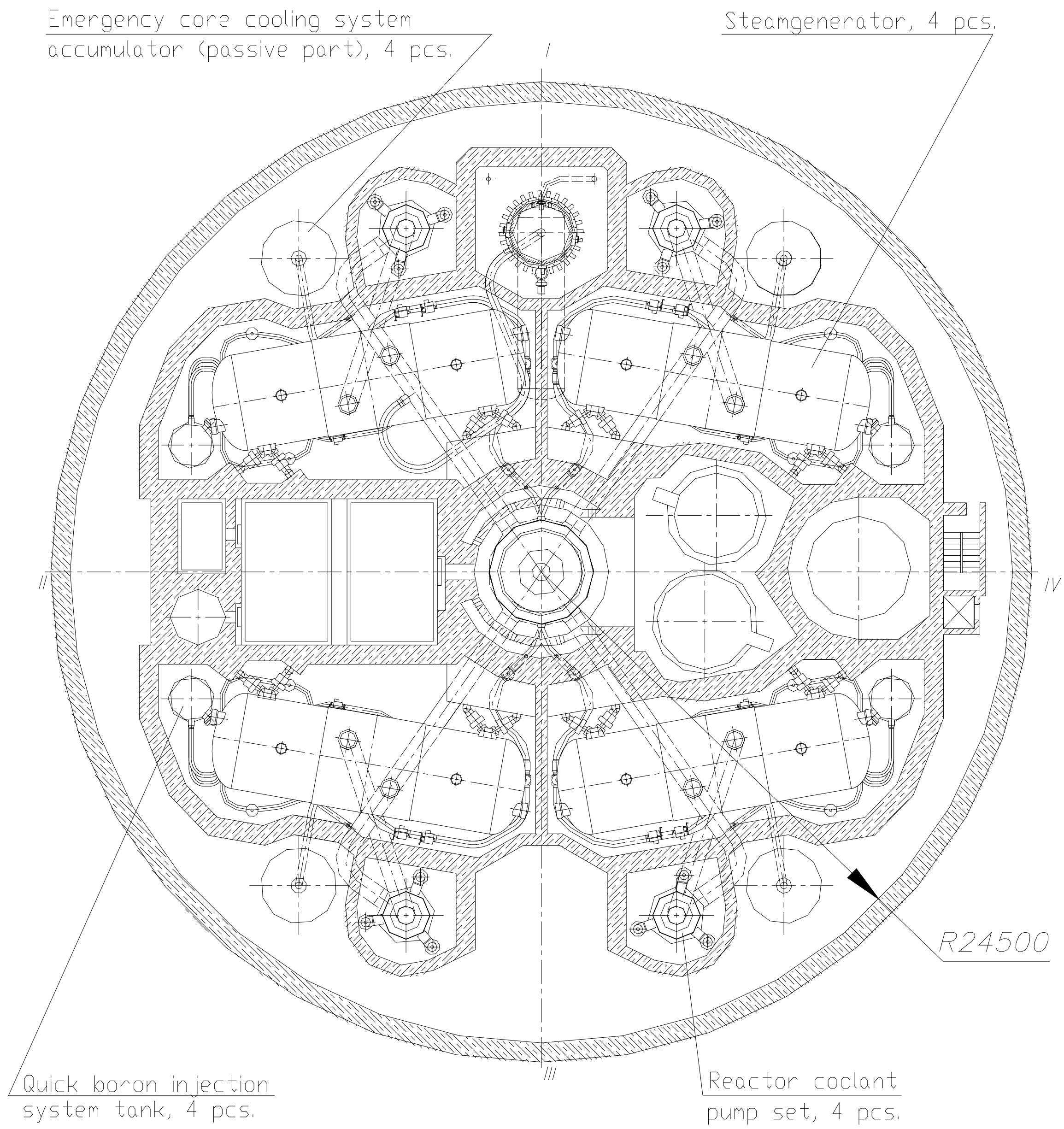


FIG. 4.15-3. Layout of the RP V-448 equipment (cross section)

4.15.2.2 Reactor core and fuel design

The reactor core is used for heat generation and transfer from the FA fuel rod surface to coolant during the design service life without exceeding the permissible limits of fuel rod damage.

The reactor core design consists of 241 FAs wherein the CPS AR are arranged in accordance with the core cartogram in Fig. 4.15-5.

The V-1500 FA is an evolutionary progress of the AFA sheathless design used for commercial WWER-1000. The CPS AR and absorbing elements also have solutions approved on the prototypes, in particular the combined (boron carbide + dysprosium titanate) absorber is used.

The FA design provides the possibility for free thermal and radiation expansion of its components. The FA radial expansion as a function of temperature and irradiation does not exceed a gap between the fuel assemblies in the core. A margin of the spring travel available in the FA cap provides compensation for the technological tolerances and differences of temperature expansions between the FA components and the reactor internals as well as damping of CPS CR drop (CPS AR and CPS drive extension shaft engaged with it). Compression of FA through the movable cylindrical shell of the FA cap and spring unit prevents the fuel assembly from lifting in case of pressure differential.

The fuel rod consists of the cladding filled with the pellets of sintered uranium dioxide UO_2 and sealed with the plugs. The fuel rod plenum is filled with pressurized helium.

The CPS AR are designed for control of nuclear reaction in the core, keeping of power at the assigned level and its change over from one level to another, axial power field flattening, prevention and suppression of xenon oscillations. The CPS AR consists of the absorbing element cap and suspender. The CPS AR cap is a sleeve with the console ribs in which there are the holes for the absorbing element suspender.

The absorbing element consists of the cladding filled with absorbing material and sealed with the end pieces by means of welding. Boron carbide (B_4C) and dysprosium titanate ($Dy_2O_3 \cdot TiO_2$) are used as the absorbing material.

4.15.2.3 Fuel handling and transfer systems

The fuel handling systems include the handling and storage system both for the fresh and spent fuel and the refuelling control system. The fuel handling systems include:

- Spent fuel cooling pond;
- Equipment for refuelling, fuel storage and transportation, CPS AR within the reactor-fuel pond and for FA inspection;
- Defective assembly detection systems and equipment for storage of the defective fuel assemblies.

The main safety functions of the fuel handling systems are:

- Nuclear safety assurance (effective coefficient of neutron multiplication shall not exceed 0,95);
- Avoidance of cargo movement above the stored fuel, if it is not a part of the hoisting and fuel handling devices;
- Stability of the equipment at SSE and other natural phenomena;
- Avoidance of FA drop and uncontrolled movement of the mechanisms at SSE and in case of loss of power supply.

To detect the defective fuel assemblies there are two kinds of leak-check for fuel rod claddings - on the operating reactor and on the shutdown reactor. Leak-check of fuel rod claddings on the operating reactor allows to define leaky fuel rods, their number and degree of loss of integrity by the primary coolant.

On the shutdown reactor provision is made for inspection during FA transportation in the fuel handling machine mast as well as inspection by the “bottle” method used to analyze water samples taken from the defective assembly detection system (DADS) circuit in whose bottle the controlled FA is placed.

4.15.2.4 Primary components

Reactor

Schematic diagram of the reactor is shown in Figures 4.15-4 and 4.15-5. The V-1500 (V-448) reactor design has been developed proceeding from the experience gained in design and operation of a series of WWER type reactors as well as with account for developments executed by OKB "Gidropress" for the last 10 years.

The reactor V-1500 like the reactors V-1000 has two-row arrangement of the nozzles. The materials for reactor fabrication are chosen with account for the required physical-and-mechanical characteristics, adaptability to manufacture, weldability and operability during operation throughout the specified service life equal to 50 years. The reactor consists of the pressure vessel, upper unit, core, internals, CPS drives.

Reactor vessel

The reactor pressure vessel consists of the following main parts: flange, upper and lower shells of the nozzles area, cylindrical shells and elliptic bottom welded between each other by the circumferential welds. The vessel has 2 rows of nozzles with the inner diameter 850 mm (4 nozzles in each row) and 4 ECCS nozzles Dnom 300 (2 nozzles in each row). On the vessel there is the instrumentation nozzle with inner diameter of 300 mm.

The vessel is installed and fixed on the supporting ring mounted on the concrete pit supporting truss. The vessel is protected against seismic loads by means of the thrust ring installed on the flange.

Upper unit

The upper unit consists of elliptic top head with the nozzles and the housings for CPS drives arrangement (118 pcs), traverse, metalwork, nozzles for ICID (28 pcs), air vent nozzle and level gauge nozzle.

Reactor internals

The reactor internals consist of core barrel, protective tube unit and core baffle. The core barrel is intended for arrangement of the reactor core. The core barrel design is in principle similar to that of V-1000 core barrel. The protective tube unit is intended to protect the CPS drive components against coolant effect at the core outlet and to provide FA fixing on coordinates in the core. The core baffle is used to protect the vessel against effect of neutron flux onto the reactor vessel walls.

CPS drive

The CPS drive whose prototype is CPS drive ShEM-3 of V-1000 reactor is accepted as the CPS AR actuating mechanism.

The drive consists of the following main parts: housing, electromagnet unit, motion unit, extension shaft, position indicator.

Main coolant pipeline (MCP)

The MCP makes use of the pipelines Dnom 850 as applied in WWER-1000 reactor plant. The MCP is of the standardized typical size with maximum scope of pre-fabrication. The MCP is designed with account for "leak before break" concept to be implemented in the reactor plant design.

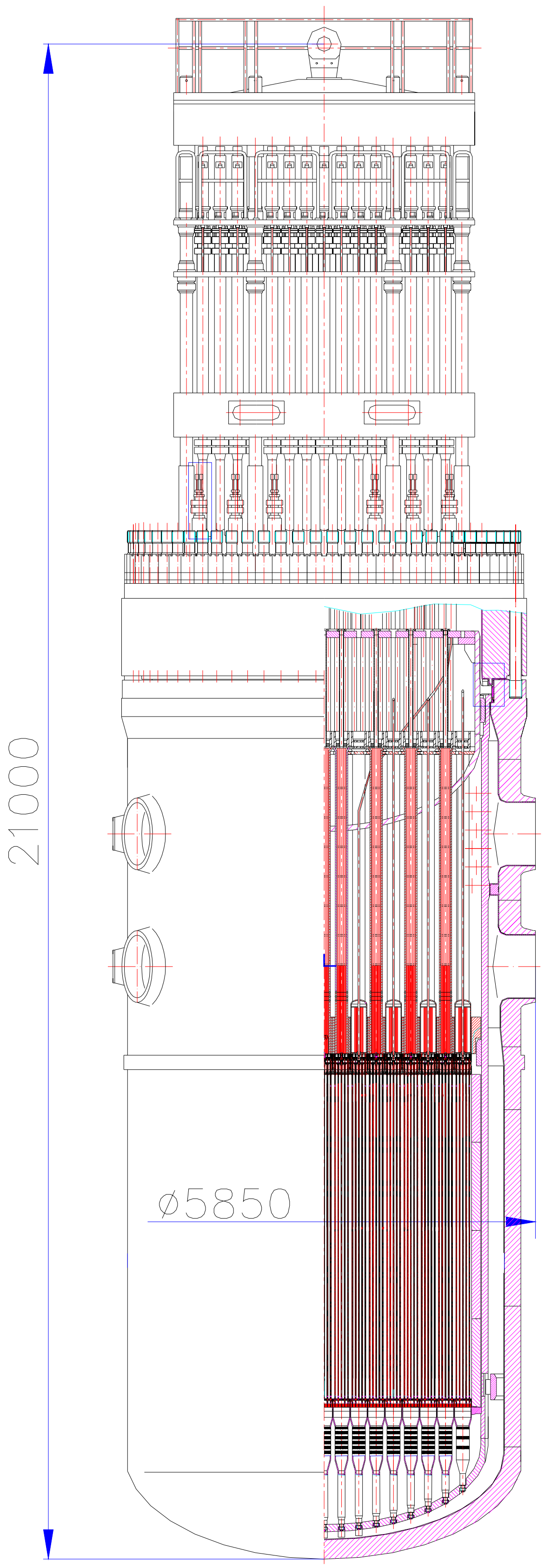


FIG. 4.15-4. Reactor. Longitudinal section.

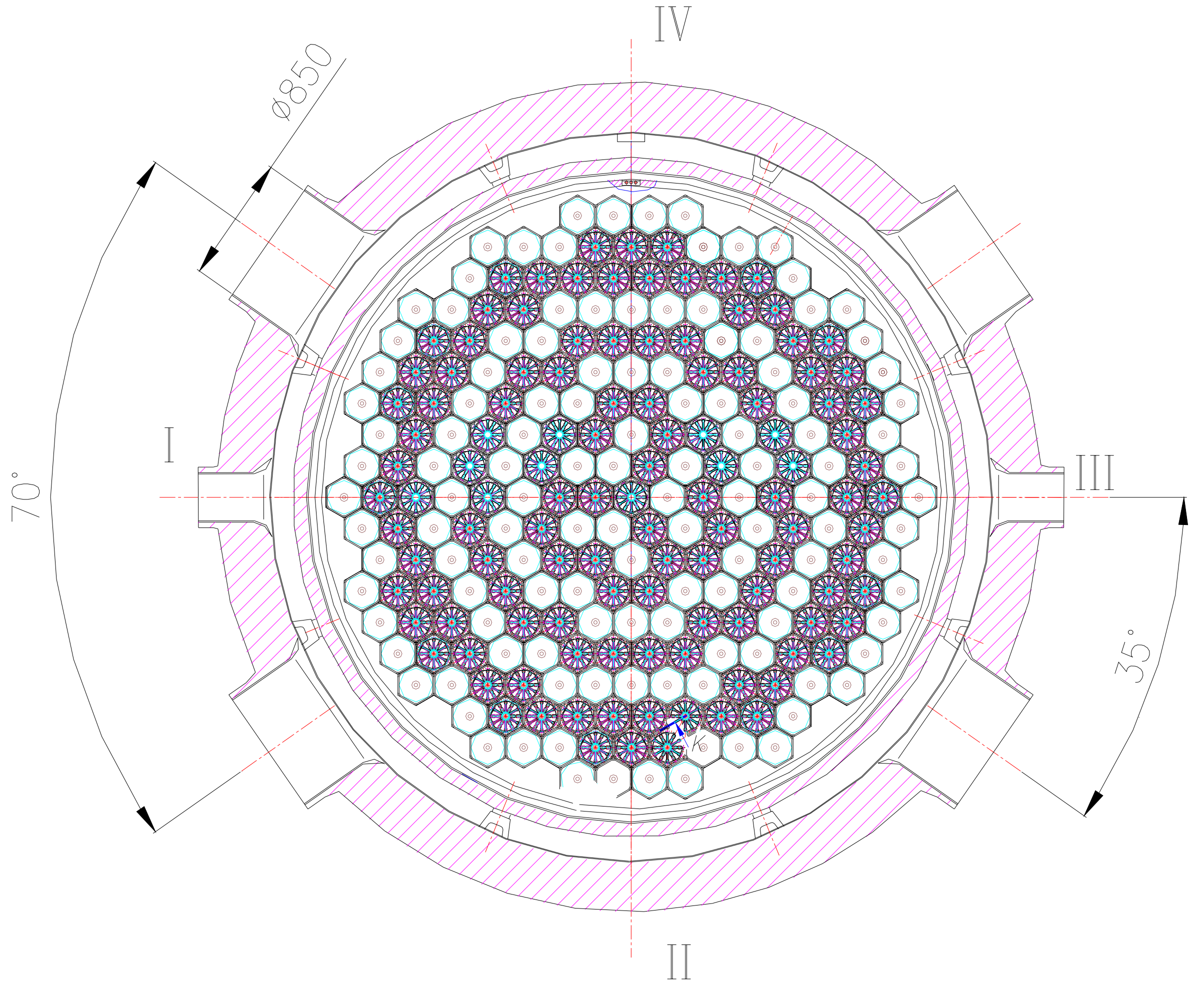


FIG. 4.15-5.   Reactor. Cross section.

Reactor coolant pump set (RCPS)

The RCP set is designed for coolant circulation in the primary circuit. In development of RCPS for WWER-1500, there were used the structural solutions of separate units (seal, main thrust bearing, auxiliary systems, coupling etc.) applied in the pump sets with operational experience gained at operating NPPs with WWER-1000. The RCP set design (Figure 4.15-6) makes use of the pump spherical casing; the pump bearings and electric motor are made with the water lubricant to exclude application of oil in the pump set The RCP set design is realized as follows: suction - from below, head – from the side.

Steam generator and pressurizer

The horizontal steam generator (Figure 4.15-7) is used in WWER-1500 design. The steam generator design is based on domestic experience in development, fabrication and operation of the horizontal steam generators with distribution perforated sheet and in-line arrangement of the heat-exchanging surface. Service life of steam generator is 50 years.

The pressurization system includes: pressurizer (Figure 4.15-8), relief tank (Figure 4.15-9), connecting pipelines with valves, as well as safety devices. The pressurizer is similar to the WWER-1000 pressurizer of increased height and represents itself a cylindrical vessel with elliptic bottoms installed vertically. The relief tank is aimed to collect the steam-gas mixture from the primary circuit during operation and represents itself a horizontal vessel with two elliptic bottoms and internals.

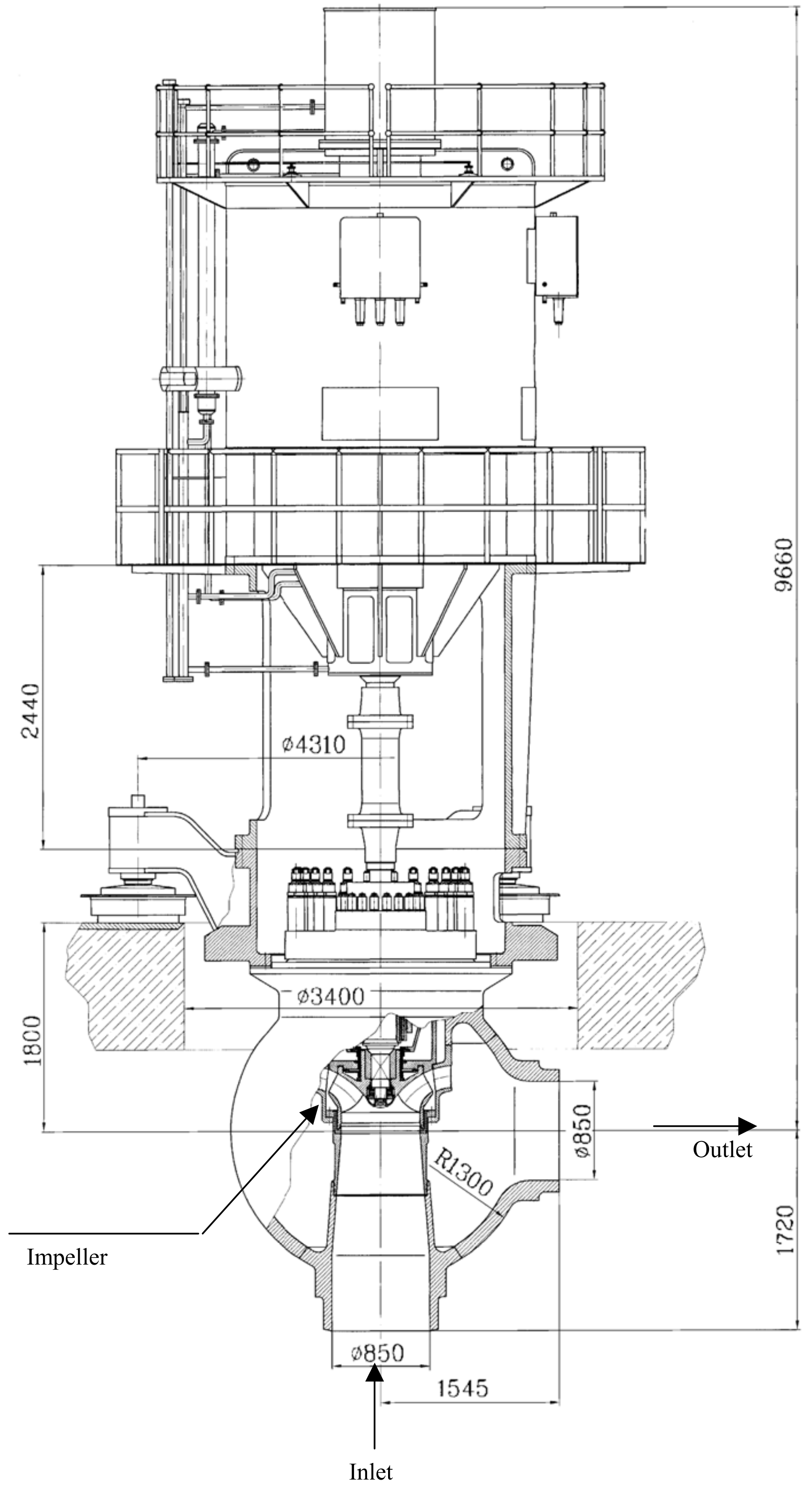


FIG. 4.15-6. RCP set

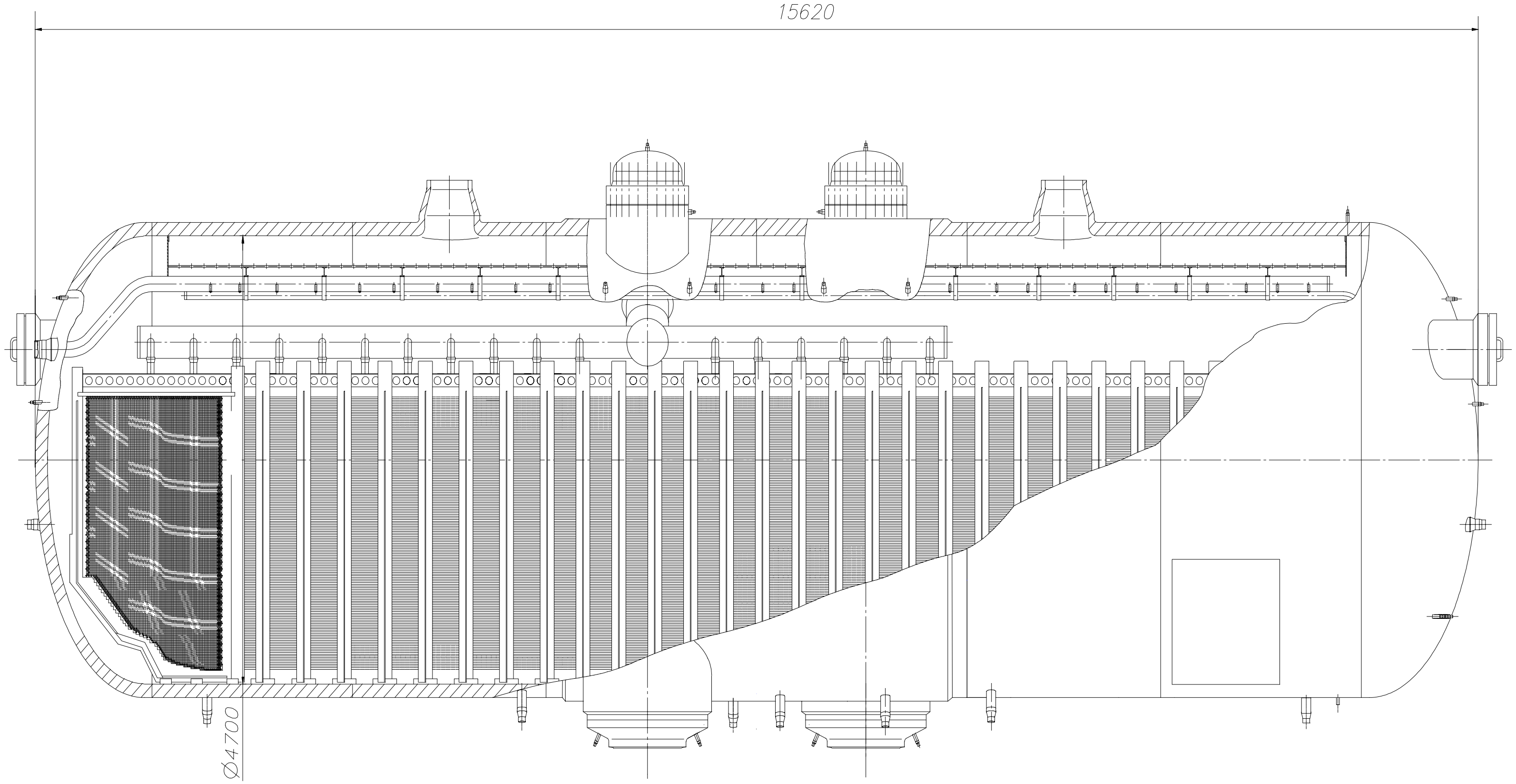


FIG. 4.15-7. Steam generator

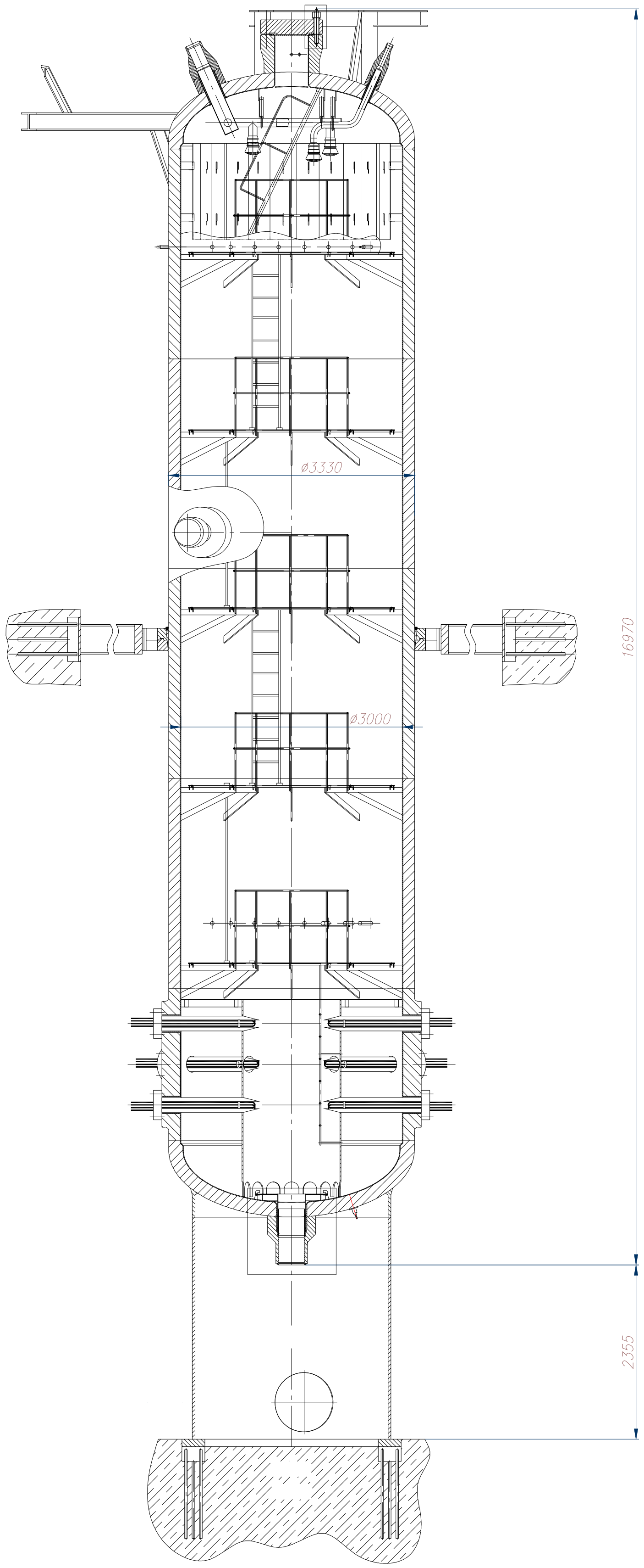


FIG.4.15-8. Pressurizer

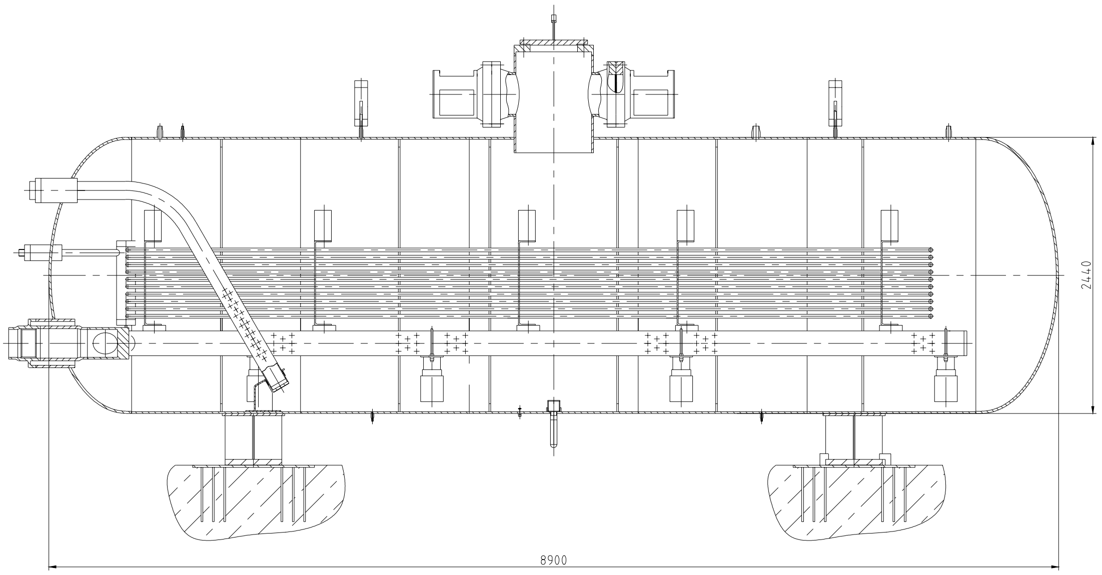


FIG. 4.15-9. Relief tank

4.15.3. Description of the turbine generator plant systems

No information provided.

4.15.4 Instrumentation and control systems

The instrumentation and control systems (I&C) are designed to provide monitoring and control of the reactor plant systems in the automatic and automatized modes. The I&C systems are constructed with account for the requirements of IAEA and Russian regulatory documents on safety.

The main functions performed by the I&C are as follows:

- Emergency and preventive protection of the reactor;
- Automatic reactor power control and power distribution management in the core;
- Generation of signals for start-up of the safety systems;
- Regulation of the RP main parameters;
- Technological protections and interlocks of the RP equipment;
- Diagnostics of the RP equipment;
- Monitoring, signaling and recording of the RP parameters;
- Diagnostics of the I&C engineered features,
- Automated control of the operational conditions.

Automatic control under the accident conditions is performed by CPS and ESFAS ensuring reactor scram and start –up of the process safety system .

The control and protection system alongside with other functions provides decrease of reactor power when the operating conditions are changed by action of PP-1, PP2, APP, and also emergency protection of the reactor by de-energizing the drives of all control rods and their drop by gravity to the extreme lower position during generation of the scram signals.

The PP-1 is performed when the controlled parameters reach the corresponding setpoints or when the RP operating equipment is changed, and also from the MCR operator key. It provides automatic decrease of reactor power by means of one-by-one movement downwards of the control rod groups starting from the operating one with the working velocity.

The PP-2 is performed when the controlled parameters reach the corresponding setpoints by means of prohibition of control rod movement upwards. The movement downwards in this case is allowed.

Action of PP-1 and PP-2 stops with disappearance of the initial-cause signal.

The APP is performed with disconnection of the working equipment and at corresponding RP parameters by means of drop of one predetermined group (both in the automatic mode and from the MCR key).

The Emergency Protection signal (EP) has a priority before other kinds of the protections and control signals. The EP action shall be carried to the end even in case of disappearance of the initial-cause signals.

The CPS is based on two independent sets of EP-PP equipment (each of which provides reliable reactor scram or restricts its power under all anticipated operational occurrences) and the I&C equipment for reactor power and reactivity control during normal operation. Each set of the CPS equipment provides generation of the control and protection signals via three independent channels for each technological parameter, by which it is necessary to perform protection.

The automatic control systems are intended to keep the main technological parameters within the permissible limits or to change them by a certain law under all possible normal operating conditions and faulty conditions without self-excited oscillations caused by interactions of the controllers in the course of regulation.

The RP main technological processes are controlled from the main control room under all operating conditions of NPP. For emergency shutdown of the reactor plant in case of MCP damage provision is made for the emergency control room to be located separately from the MCR. Remote control of the RP equipment, devices and valves from ECR is carried out individually.

4.15.5 Electrical systems

No information provided.

4.15.6 Safety concept

4.15.6.1 Safety requirements and design philosophy

The NPP meets the requirements for safety if its radiation effect on the personnel, population and environment during normal operation, anticipated operational occurrences, including the accidents, does not result in exceeding the assigned irradiation doses of the personnel and population, standards for excursions and drops, content of radioactive substances in the environment and also if it is limited during the beyond design-basis accidents.

The NPP safety is ensured due to the sequential physical barriers on the way of potentially possible propagation of ionizing radiation, radioactive substances into the environment and the system of the technical and organizational measures on protection of the barriers, the keeping up of their efficiency and those on population protection.

The system of physical barriers includes:

- Fuel matrix;
- Fuel rod cladding;
- Leak-tight reactor coolant primary circuit including the leak-tight heat-exchanging surface of steam generators and heat exchangers, vessel designs, pipelines;
- containment, reactor concrete pit and corium catcher. The barrier protection is performed by the safety system that ensures the fulfillment of the main (critical) safety functions with the help of the channels being independent from each other and acting on the basis on various principles - active and passive.

The specified principles cover such critical safety functions as:

- Fast bringing the reactor into subcritical state;
- Decay heat removal from the reactor including the cases with the coolant leaks;
- Maintaining of the coolant material equilibrium in the reactor vessel;
- Cooling of double concrete containment under the accident conditions;
- Suppression of hydrogen releasing into the containment during accidents;
- Localization and cooling of fuel melt during severe accidents.

The deterministic approach

The technical purpose of safety is application of the engineering and organizational means in order to ensure the measures directed at preventing the accident at the NPP, limit their radiological consequences, ensure “practical impossibility” of the accident with the large radiological consequences.

The achievement of the technical purposes of safety is ensured by solving the following tasks:

- Decrease in frequency of initiating events that disturb normal operation due to increasing the quality of the equipment and systems and their operation;
- Implementation of the complex of engineering systems and means for overcoming the design and beyond design-basis accidents including ATWS and blackout;
- Decrease in probability of severe core damage including that on the shutdown reactor (less than 10^{-6} per reactor-year), due to implementation of the complex of the active and passive systems, properties of inherent safety;
- Decrease in probability of limiting, emergency excursion of radioactive products into the environment up to the level less than 10^{-7} per reactor-year due to implementation of the passive safety systems, increase in reliability of other accident management systems including the localizing means and systems including the double containment.

The limits of fuel rod damage for normal operating conditions (operational limit), anticipated operational occurrences (safe operation limit) and the reference isotope radioactivity levels related to the this are set in the design according to the requirements for the valid regulatory documents.

During the design-basis accidents the maximum design limit of fuel rod damage specified by the valid normative documents shall not be exceeded and the number of the failed fuel rods shall be assessed.

The safety assessment is performed by comparing the results of analyses with acceptance criteria, and also the analyses of radiological consequences.

During the analysis and justification of safety two complementary methods are used: deterministic and probabilistic.

Within the framework of the deterministic and probabilistic analysis of designs, systems and components of the NPP their conformity to rules and regulations being valid in atomic power engineering is justified.

The analysis shall show that the response of the RP and its safety systems corresponds to the previously defined requirements concerning as the system characteristics so the accident consequences.

The probabilistic analysis justifies the deterministic analysis as regards a choice for consideration of the beyond design-basis accident scenarios, assessment of efficiency of solutions on safety and as a whole ensures the balance of deep-in-depth protection and demonstrates the fulfillment of the probabilistic safety indices that are regulated by the regulatory documents.

Decrease in risk

The main principle of the reactor plant safety assurance is assumed to be the principle of an internal inherent safety on the basis of natural feedbacks and processes.

The RP internal inherent safety is expressed in ability to limit a sequence of the initiating events and their consequences within the design safety criteria limits without participation of the personnel and external help for a long time. This time shall be used by the personnel for evaluating the situation and performing out the corrective actions.

The criteria of the inherent safety level are duration of a permissible "period of non-interference" in the various situations (not less than 30 min.), sluggishness of emergency processes.

These properties of internal inherent safety are caused by:

- Volumetric power density of the core and linear power density of the fuel rods being decreased as compared with the high-power WWER in operation;
- Properties of self-restriction of the core power due to the negative reactivity coefficient of fuel temperature, coolant temperature, power under the entire operating range of the parameters;
- Implementation of the "leak before break" concept as applied to the pipelines of the primary and secondary circuits;
- Heat-engineering and strength margins;
- Increase in the margins of water in the emergency core cooling systems (including usage of water of ponds and tanks of other systems);
- Application of the passive BDBA control systems.

The properties of the reactor internal inherent safety are directed at self-restriction of power and self-shutdown, limitation of reactor pressure and temperature, heat-up rate, scale of loss of integrity of the primary circuit and outflow rate, scales of fuel damage, preservation of reactor vessel integrity during the severe accidents.

The strategy of prevention of deviations, disturbances and accidents is of priority for the designer.

4.15.6.2 Safety systems and features (active, passive and inherent)

Safety system configuration

Depending on the leak size the design-basis accidents with loss of integrity of the primary circuit proceed with various rate of loss of mass and energy of the coolant and decrease in the pressure. To compensate the coolant leaks the design provides for water supply from the ECCS components in a certain sequence.

In case of occurrence of the initiating event related to primary coolant leak, the reactor CPS ensures the emergency shutdown of the reactor by de-energizing the drives of all control rods and their dropping under the own weight action up to the extreme lower position and the automatic start-up of pumps of the emergency and planned cooldown system is performed and the water supply into the core is ensured. Owing to the on-going outflow of the coolant the primary pressure is decreased and under pressure of 4,4 MPa the core flooding from the hydrotanks of the first step is ensured.

While draining the tanks the reactor core is cooled down by water supplied into the primary circuit by the emergency cooldown pumps. At the first stage of the accident the pumps draw water from the cooling pond. After decreasing the pond level up to the level being sufficient to ensure the biological shielding function, water intake from the pond stops and starts from a sump wherein the water is accumulated. This water is outflowing into the leak from the primary circuit; in this case for compensation of the water being evaporated from the pond, the part of water from the pump is supplied through the auxiliary ejector into the pond. In case of loss of NPP power when the initiating event occurs the water supply from the pumps is delayed for the time being necessary for start-up of diesel-generator and start-up of the pumps. The characteristics of the hydrotank of the first step are chosen with consideration of this delay. The emergency and planned cooldown system is functioning during the entire period of the accident and during the post-accident period. In case of the initiating events “LB LOCA” and “SB LOCA” the non-condensing gases and coolant from under the reactor top head and from the SG header through EGRS are released.

In case of blackout for a long time both as under the condition of preservation of leak-tightness of the primary circuit so under the condition of occurrence of leaks in the primary and secondary circuits the PHRS starts operating. During loss of integrity of the primary circuit the PHRS is working together with the system of hydrotanks of the second step that ensures for the required coolant level in reactor vessel to be maintained in the reactor vessel within 24 hours in order to remove the heat from the core using the PHRS. The passive spillage from the tank is ensured by draining the boron solution from the tanks into the reactor in time the discharge pipelines from the different elevations in the tank under the action of hydrostatic head from the initial primary pressure of 1,5 MPa.

Safety systems for protection and cooling of the core that fulfil the main RP safety functions include:

- Reactor CPS;
- Quick boron injection system (QBIS);
- Emergency boron injection system (EBIS);
- System of emergency and planned cooldown of the primary circuit and cooling of the cooling pond (CP);
- Core passive flooding (CPF) (ECCS tanks of the first step);
- Additional system for core passive flooding (ASCPF) (ECCS tanks of the second step);
- Passive heat removal system (PHRS);
- SG emergency cooldown and blowdown system;
- Emergency gas removal system (EGRS).

Reactor CPS

The system ensuring change-over of the core into the subcritical state and restraining it in this state, is the CPS reactor electromechanical system. The purpose of the CPS is given section 4.15.4.

In combination with the core cooling systems and the properties of the reactor inherent safety the efficiency of the emergency protection is sufficient for change-over of the reactor into the subcritical state during the accidents up to temperature of 100 ^{o}C in case of current pre-accident boric acid concentration in the primary coolant and its maintaining in this state as long as one may like.

Quick boron injection system

QBIS is intended to change over the reactor into subcritical state under the conditions being accompanied by failure of EP to operate (beyond design-basis accidents such as ATWS). The change-over of the reactor into subcritical state is ensured by insertion of negative reactivity due to increasing boron concentration in the primary coolant from four tanks being independent from each and connected to the MCP using the connecting pipelines. Connection is performed at the RCPs head and suction. On the connecting pipelines the normally closed quick-acting isolation valves are placed.

While appearing the signal for scram and failure of the EP to operate the quick-acting shut-off valve opens and due to RCP pressure differential the boric acid solution from the QBIS tank enters into the primary circuit. Boric acid solution is supplied from the QBIS tank also in case of RCP set shut-down (due to its coastdown).

Emergency boron injection system

EBIS is intended for emergency supply of the high-concentrated boric acid solution into the reactor core by the pumps during the accidents with maintaining the high primary pressure, and also for decrease in the pressure by cold water injection into the PRZ steam space. The pressure pipelines of the pumps are connected with the cold legs of the circulation loops and with the PRZ upper part.

The system of emergency and planned cooldown of primary circuit and cooling of the cooling pond.

The system of emergency and planned cooldown of primary circuit and cooling of the cooling pond is the ECCS active part and is intended to fulfil following safety functions during design-basis accidents:

- Water supply into the reactor core under the conditions related to loss of tightness of the primary circuit;
- Reactor core cooling and subsequent long-term decay heat removal during the accidents related to loss of tightness of the secondary circuit or failure of the systems of normal heat removal through the secondary circuit;
- Water supply into the cooling pond to ensure fuel cooling during the accident conditions.

Besides the system is intended to fulfil the following functions of normal operation:

- Planned cooldown of the primary circuit during RP shut-down and decay heat removal in case of overloading the core including the cases when the RP equipment is repaired with decreasing the reactor coolant level up to the “cold” legs nozzle axis without core unloading;
- Heat removal from the spent fuel cooling pond.

The system is made as two independent subsystems, each of which consists of two channels respectively.

Each channel of the system includes:

- High pressure pump,
- Ejection plant from the delivery side,
- Emergency cooldown heat exchanger,
- Pipelines and valves.

Core passive flooding system

The core passive flooding system includes the system of the ECCS tanks of the first step and the system of the ECCS tanks of the second step.

The ECCS of the first step is a traditional system being the part of the reactor plant structure in all the designs of the WWER-1000 NPPs and in case of the initiating events with loss of the primary coolant the ECCS is to ensure the replacing of loss of the coolant and core cooling.

The system of the ECCS tanks of the second step is an additional system and intended for passive supply of boric acid solution into the reactor core under the primary pressure of 1,5 MPa and less in order to flood the core during LOCAs under the conditions of blackout for a long period. The system consists of four groups of the accumulating tanks with boric acid solution of 16 g/kg. The tank system shall ensure realization of self-profiling of water flow rate entering into the reactor from the ECCS tanks with decreasing the flow rate as the parameters decrease and intensity of the outflow of medium from the primary circuit decreases.

Passive decay heat system

The PHRS is intended for long-term reactor decay heat removal under the conditions of loss of non-emergency a.c. power to the station auxiliaries as on condition of keeping the tightness of the primary circuit, so on condition of occurrence of leaks in the primary and secondary circuits. In case of leaks in the primary circuit the system works together with the tanks of the second step.

The PHRS is the closed circuits of natural circulation for decay heat removal from reactor due to steam condensation being removed from the SG and returning the condensate into the SG. Each circuit includes four heat-exchanging modules, steam-condensate path pipelines with the valves, path of air ducts (boxes), supplying and discharging air, with the gates. Decay heat removal is performed by the system directly to the outdoor air in the heat exchanger - condenser.

SG emergency cooldown and blowdown system

SG emergency cooldown and blowdown system is an active system of the decay heat removal to the ultimate heat sink and it is intended for:

- Reactor core decay heat removal and cooldown of the reactor plant under the accident situations related to loss of power or loss of possibility of normal heat removal over the secondary circuit including leaks of steam lines and feedwater pipelines of the SG;
- Reactor core decay heat removal and cooldown of the reactor plant under the accident situations related to unsealing of the primary circuit including break of the RCC pipelines (through the unaffected loops).

Besides, the system is intended for supplying the blowoff water of the steam generators for purification and its returning under the normal operating conditions.

The system consists of two independent subsystems, each of which includes two channels. Each of four channels is capable to fulfil the safety functions imposed on the entire system as a whole.

The channel of the system includes the pipeline of connection with a steam generator steam line, regenerative heat exchanger, emergency cooldown heat exchanger (process condenser) cooled with the intermediate circuit water, pump and pipeline of returning the condensate into the steam generator.

Emergency gas removal system

EGRS is intended for removal of non-condensing gases released from the coolant in the upper points of the primary circuit during the accidents related to decreasing the primary circuit parameters in order to exclude the explosive concentration and explosions of hydrogen, and also for prevention of failure of the primary coolant natural circulation. Such points are as follows: SG headers, reactor top head, pressurizer.

Under the normal operating conditions the system ensures air removal while filling the primary circuit.

The emergency gas removal system consists of collector Dnom 65 mm with the upper sections of the primary circuit being connected to the latter: steam generator headers, reactor top head, upper part of the pressurizer. The main header is connected to the relief tank of the primary circuit pressurizing system and to the containment volume to ensure the possibility of emergency gas removal in case when the pressure in the relief tank prevents the emergency gas removal.

4.15.7 Plant layout

No information provided

4.15.8 Technical data

Main consumer data		
Nominal thermal power of reactor	4250	MW
Assigned service life of main equipment	50	years
Assigned service life of reactor vessel	60	years
Main technical characteristics		
The number of circulating loops	4	pcs
Inside diameter of MCP	850	mm
Nominal steam capacity	8250	t/h
Steam pressure at the SG outlet	7,34	MPa
Feedwater temperature (nominal)	230	^oC
Steam humidity at the SG outlet, not more than	0,2	%
Nominal coolant flow rate through the reactor	108400	m^3/h
Coolant pressure at the core outlet	15,7	MPa
Coolant temperature at the reactor inlet	297,6	^oC
Coolant temperature at the reactor outlet	330	^oC
Reactor core		
Core height (in cold state)	4200	mm
Fuel mass (UO_2 + Gd)	~141	t
Maximum linear power density of fuel rod	340	W/cm
Specific power density per unit of volume	87	kW/l
Height of FA	4900	mm
Shape of breakdown of the core cells	triangular	
The number of FAs	241	pcs
The number of fuel rod (Gd-fuel rod) in FA	306	pcs
The number of guiding channels for AR	24	pcs
The number of guiding channels for ICID (central tube)	1	pcs
The number of spacing grids	17	pcs
Average enrichment of ^{235}U make-up fuel	4,4	%

Average burn-up of fuel	55÷65	$\frac{MWd}{kgU}$
Material of fuel rod cladding	Zirconium alloy	
Outside diameter of cladding	9,1	mm
The number of CPS CR	118	pcs
The number of AR in CPS CR	24	pcs
Absorbing material	$B_4C + Dy_2O_3TiO_2$	
Reactor vessel		
Inside diameter (in the core area)	4960	mm
Total height (with a top head)	15,0	m
Material	15Х2НМФА	
Design pressure	17,64	MPa
Design temperature	350	^oC
Mass without top head	430	t
Fluence of neutrons for service life of 60 years:		
- with E>1 MeV;	$1,10 \cdot 10^{19}$	n/cm^2
- with E>0,5 MeV	$1,45 \cdot 10^{19}$	n/cm^2
Steam generator		
Type	Horizontal	
Quantity	4	pcs
The number of heat-exchanging tubes (16x1,5 mm)	15120	pcs
Inside diameter of the vessel	4,700	m
Vessel length	15600	mm
Mass (without working medium and supports)	530	t
Material of vessel and the primary-side collector	10ГН2МФА	
Material of heat-exchanging tubes	08Х18Н10Т	

RCP set		
Type	centrifugal	
Quantity	4	pcs
Flow rate	27100	m^3/h
Head	0,8	MPa
Power consumed during operation with cold water	11000	kW
Power consumed during operation with hot water	7400	kW
Frequency of rotation	1500	rev/min
Pressurizer		
Complete volume	103	m^3
Inside diameter	3,000	m
Material	10ГН2МФА	
Mass without fasteners	244	t
Relief tank		
Complete volume	36	m^3
Material	08Х18Н10Т	
Mass	16,2	t
ECCS tanks		
Quantity	4	pcs.
Water volume in the tank	50	m^3
Boric acid concentration in the tanks	16	g/kg
Pressure of actuation, less than	4,4	MPa
Quick boron injection system		
The number of tanks	4	pcs.
Design pressure	17,64	MPa
Design temperature	350	^oC
Volume of solution in the tank	11	m^3
Temperature of boric acid in the tanks	55	^oC
Concentration of boric acid in the tanks	39,5 – 44,5	g/kg

Passive heat removal system		
The number of channels	4	pcs
Power of one channel	28,3	MW
The number of heat exchangers in one channel	4	pcs.
Temperature of cooling	from -40 to +50	^oC
ASCPF tanks		
Quantity	8	pcs.
Complete volume of one tank	180	m^3
Concentration of boric acid in the tanks	16	g/kg
Pressure of actuation, less than	1,5	MPa

4.15.9 Measures to enhance economy and maintainability

The future plants shall be safer than the existing ones, and this can be achieved using the safety systems based on a passive principle, which, for some designs, is a preferable alternative for providing a reliable fulfilment of safety functions.

In the design of WWER-1500 RP, it was decided together with passive safety systems to use traditional active systems performing the same safety functions. Such a decision complicated the design a little. However, for general simplification of the design and to increase a functional reliability of the active parts of safety systems, a principle consisting in combining safety functions and normal operation functions was assumed. In this case, these systems were designed to transfer the operating channels to the mode of fulfilment of safety functions either without commands of activation and switching mechanisms at all, or with a minimum number of switchings only of the valves. In this case the number of commands for switchings, as the analysis showed, is much fewer than that taking place during start-up of traditional safety systems which are not used for performing normal operation functions. Development of the principle of combining of functions allowed to assume simple technological solutions, providing fulfilment of a number of safety functions actuating successively one after another in the course of the accident, by one set of mechanisms (for example, water supply into the core at high and low pressure). Such solutions exclude a necessity of additional switchings, being a source of failures, and allow reducing the number of units of equipment, pipelines, valves, interlockings and etc.

The principle of combining and solutions will allow to reduce considerably the undetected failures, to compensate the design complication mentioned above and to increase the safety level.

In developing the systems, the lay-out decisions, equipment characteristics, operating conditions, and control are assumed by analogy to NPP-92 design (WWER-1000/V-392). It allows to rely on the solutions licensed in Russia, which were made in developing NPP designs of new generation, on the containment, diagnostic systems, on justification of "leak before break" concept, on phenomena of severe accidents and control devices etc. Improvement of economic indices is also reached due to realization of the evolutionary approach. It includes:

- Direct application of the equipment and systems from WWER units, which confirmed their reliability and service life;
- Standardization and unification of process components and materials mastered within the framework of program for WWERs;
- Organization of refuelling cycle with periodicity from 10 to 24 months;
- Increase in fuel burn-up to 55-65 MWd/kgU and a corresponding decrease in specific indices of fuel consumption for electricity production.

Measures for reduction in the scope of maintenance and repair during operation and refuelling, which are provided for in the design, are important for operation and economic indices. The scope of maintenance and repair of RP equipment (which have an influence, together with optimization of fuel cycle and other measures, on an increase in the value of capacity factor), was decreased in the WWER-1500/V-448 design by:

- A reduction in the duration of RP scheduled shutdowns for maintenance and repair by means of:
  - optimization of the repair cycle structure, including an increase in the period between repairs;
  - improvement of organization for preparation and performance of maintenance and repair;
  - quality improvement of personnel training;
  - optimization of work scope;
  - application of high-efficiency means of technological equipment for maintenance and repair (automated monitoring systems, multiposition power-nut drivers, repair equipment and etc.);

  - increase in the rate of performance of fuel handling procedures (increase in velocity of fuel handling machine and polar crane in the reactor hall);
- Improvement of maintenance and repair technologies;
- Improvement of regulating documentation relating to maintenance and repair;
- Reduction in duration of unscheduled RP shutdowns in connection with the usage of up-to-date, main equipment of the reactor plant with the improved characteristics of reliability (increased service life, improved characteristics of repairability and etc.);
- Improvement of logistics;
- Improvement of repair quality;
- Improvement of information support of maintenance and repair.

All these design measures, together with measures for the unit, will allow to lower the generation cost of the produced electric power.

4.15.10 Project status and planned schedule

No information provided.

4.16 SUPERCRITICAL-WATER COOLED POWER REACTOR (TOSHIBA CORP. / HITACHI LTD. / UNIVERSITY OF TOKYO, JAPAN)

4.16.1 Introduction

Supercritical-water Cooled Power Reactor[1] (SCPR) operates above the thermo dynamic critical point of water (374°C, 22.1 MPa). The key advantages over the current generation light water reactors (LWRs) include:

1. The estimated thermal efficiency exceeds 40% due to supercritical-pressure (SCP), high-temperature steam at the turbine inlet (Fig. 4.16-1).
2. Thermal components such as heat exchangers and turbines are compact and the need for steam separation systems, re-circulation systems as well as steam generators, is eliminated (Fig. 4.16-2) because of the higher enthalpy content of the Super-critical Water (SCW) and because of no phase change in supercritical regime.
3. The R&D cost would be minimized because this technology is based on matured LWR technologies as well as matured SCP fossil power technologies (Fig. 4.16-3).
4. the good properties of super-critical water facilitates designing the moderator volume to achieve either a thermal or a fast neutron spectrum.

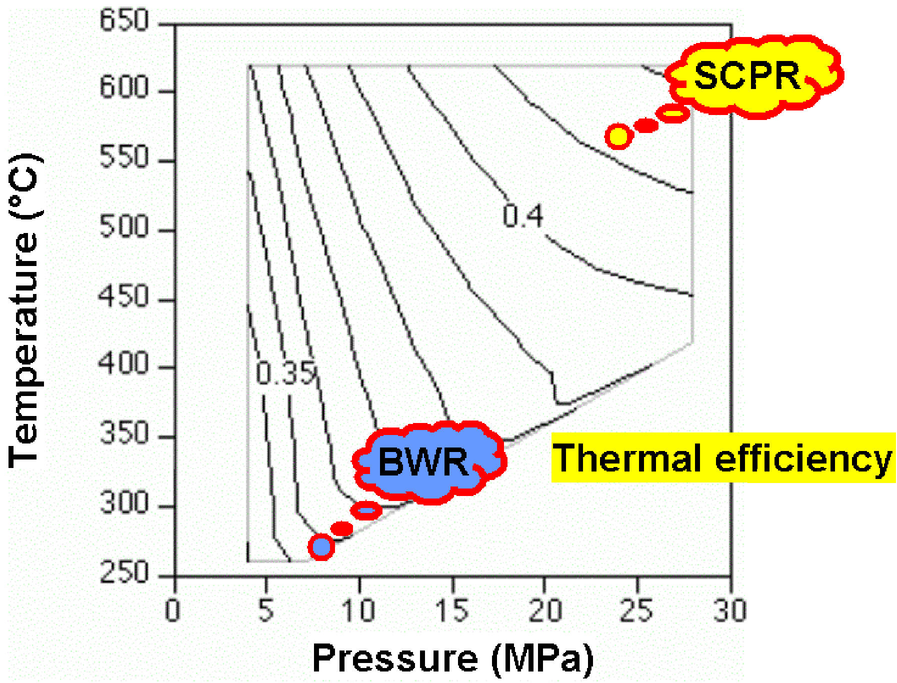
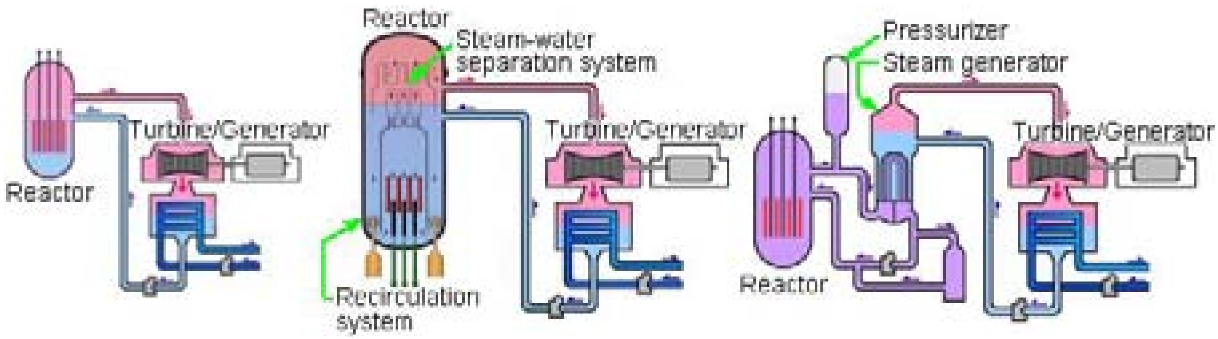


FIG. 4.16-1. Estimated thermal efficiency of SCPR and BWR [1]
(Regenerative stages: 6, Condenser pressure: 5 kPa)

[1] The US Generation-IV participants name this concept SuperCritical Water cooled Reactor (SCWR). [2]



SCPR BWR PWR

FIG. 4.16-2. System comparison between SCPR and conventional BWR/PWR [1]

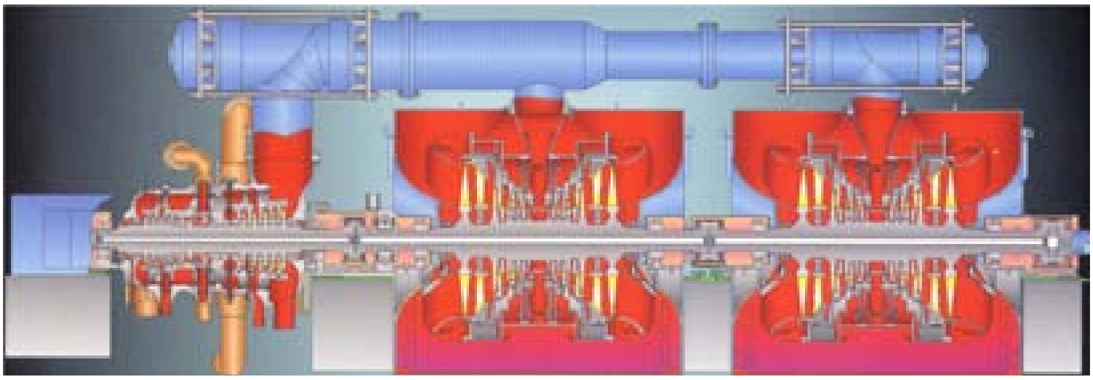


FIG. 4.16-3. Operating supercritical pressure turbine of Nanao-Ota #2 fossil power station [3]

The engineering feasibility of the SCPR as a high converter or even breeder reactor for fissile material is sufficiently promising that some development has been carried out on these concepts which could contribute to sustainable energy production for the future. SCP water is more compatible with tight lattice core than LWRs because the core coolant flow rate is smaller due to the high-coolability. Besides, negative reactivity at coolant loss can be achieved by applying zirconium-hydride layer in the core. These features facilitate the SCPR to be designed as a fast reactor.

4.16.2 Description of the nuclear systems

The following mainly describes features of SCLWR-H (1000 MWe) developed by the University of Tokyo [4][5]. Major characteristics of SCLWR-H are summarized in TABLE 4.16-I. compared with a typical LWR's characteristics. The schematic view of SCLWR-H is shown in Fig. 4.16-4. Although design options are proposed by Toshiba Co. as well as Hitachi Ltd., they are not reported in this document.

TABLE 4.16-I. MAJOR CHARACTERISTICS OF SCLWR-H [4][5]

		SCLWR-H*1	ABWR
Electric power	MW	1000	1356
Thermal power	MW	2273	3926
Efficiency	%	44	35
Core cooling		Once through	Recirculation
Reactor pressure	MPa	25	7
Core inlet temp.	°C	280	278
Core outlet temp.	°C	508	287
Core flow rate	t/s	1.2*1	14.5
Fuel material		UO_2	UO_2
Cladding material		Ni alloy*2	Zircaloy
Steam cycle		Direct	Direct
Steam pressure	MPa	24	7
Steam flow rate	m^3/s	14	60

*1 Options include a range of plant rating at 1000-1700 MWe
*2 Stainless steels are also good candidates.

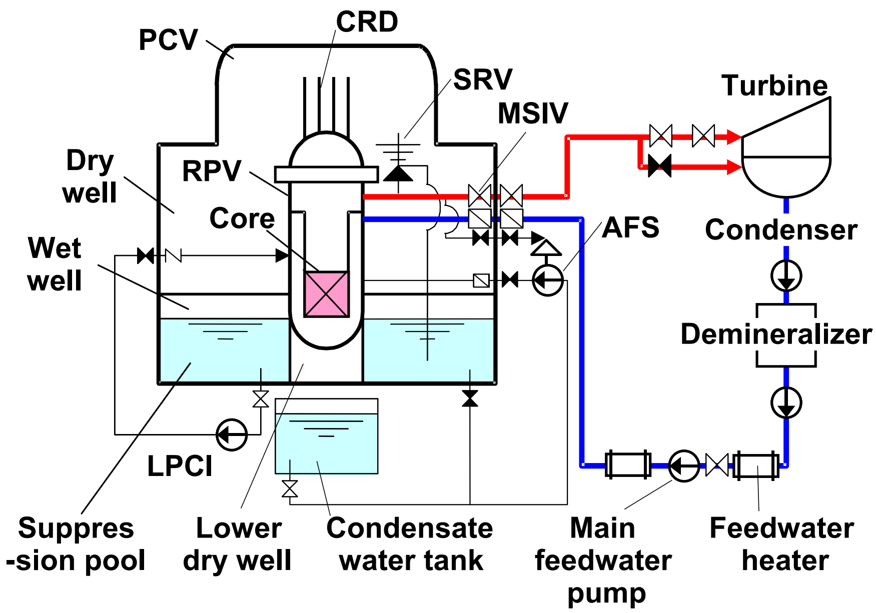


FIG.4.16-4. Schematic view of SCLWR-H [1][4][5]

4.16.2.1 Primary circuit and its main characteristics

The core is cooled by supercritical-pressure water at 25 MPa, which was chosen from the matured SCP fossil power technologies. In the SCLWR-H (1000 MWe) core, coolant is heated from 280 to 508°C at 25 MPa. The core flow rate of 1.2 t/h, which is 8% of ABWR's flow rate, removes heat from the fuel rods to suppress the rod surface temperature far below 620°C.

4.16.2.2 Reactor core and fuel design

The reactor core consists of relatively large 96 square fuel assemblies (about 301 x 301 mm) proposed by the Univ. of Tokyo (Fig. 4.16-5). The fuel assembly consists of 301 fuel rods and 36 water rods. The fuel rod mainly contains UO_2 like LWR fuels in the nickel alloy cladding. The water rods in the fuel assembly compensate small moderation resulted from the large change in coolant (water) density along the core as well as to guide the control rod insertion into the core (Fig. 4.16-5). In the water rods, cold coolant descends from the RPV top head through the control rod guide tubes located in the hot plenum. Along the coolant channels between the fuel rods or between the fuel rod and the water rod in the channel box, coolant ascends from the core bottom to the hot plenum. This arrangement facilitates neutron moderation by cold water in the water rods and core cooling by water along the coolant channels.

Other thermal hydraulic parameters are designed under the following design criteria mainly based on LWR experiences [5].

- Maximum cladding surface temperature: 620°C for nickel alloy cladding,
  (450°C for stainless steel cladding)
- Maximum linear heat rate: 39 kW/m
- Negative coolant void reactivity

The reactor is shut down by inserting the cluster type control rods from the top of the core. A preliminary evaluation shows that number of control rods is 16 per fuel assembly to achieve cold shutdown.

According to the preliminary evaluation by Toshiba, the U^{235} enrichment for the equilibrium core exceeds 6% to achieve the similar discharge burnup as current LWR fuels. This high enrichment is mainly necessary due to the relatively high neutron capture of the structure materials including the fuel cladding and the channel boxes. To ensure adequate shut down margin during the entire operation cycle, gadolinia (burnable poison) is incorporated in the fuel.

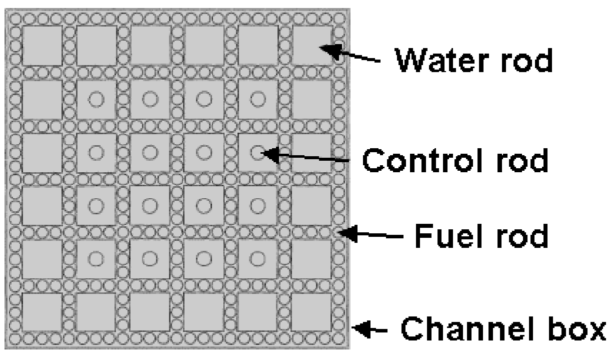


FIG. 4.16-5. Cross-sectional view of SCLWR-H fuel assembly [4][5]

4.16.2.3 Fuel handling and transfer systems

Same systems as current LWRs utilize are applicable.

4.16.2.4 Primary components

The structure of the SCLWR-H Reactor Pressure Vessel (RPV) is similar to that of PWRs, whose inner surface temperature is kept at the coolant inlet temperature. The inner diameter is about 4.3 m; the total height is 15 m; the thickness of the shell is about 0.39 m according to Toshiba's evaluation.

The structure of control rods is similar to those used for PWRs. The control rod clusters are used for primary reactivity control of SCLWR-H. The drive mechanisms are mounted on the top of the RPV. All RPV walls are cooled by the inlet coolant as in PWRs so that same materials can be used for SCLWR-H as used for PWRs.

There is no active primary component used for SCLWR-H. High-pressure cold coolant is fed by the feedwater pumps in the turbine building. No coolant circulation nor re-circulation pump is accommodated. No steam separation systems are necessary. The primary system of SCLWR-H is simple.

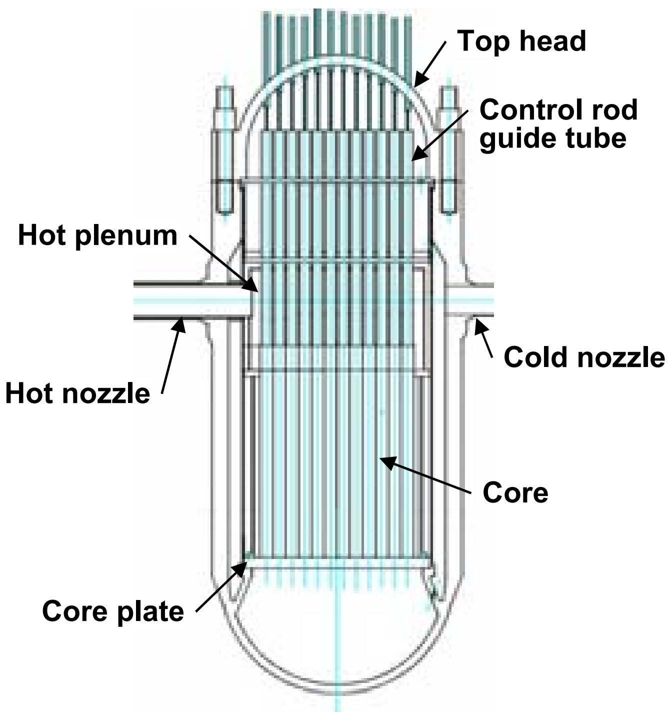


FIG. 4.16-6. Reactor pressure vessel of SCLWR-H designed by Toshiba

4.16.2.5 Reactor auxiliary systems

Residual heat removal systems (RHRs) are being optimized for SCLWR-H based on current LWR technologies. Because RHRs function only when the reactor pressure is low[1], design requirements for RHRs are similar with those for RHRs of current BWRs.

Coolant chemical control systems and purification systems are being developed based on current LWR as well as SCP fossil power technologies. Those systems suppress corrosion or stress corrosion cracking (SCC) in the entire plant.

The systems and components for refueling are not different in primary design from those installed in current LWRs. The systems for radioactive waste treatment and heating ventilating and air-conditioning (HVAC) are primarily same as those for current LWRs.

4.16.2.6 Operating characteristics

SCLWR-H is being designed to achieve better or equal operating characteristics relative to current LWRs. The major operating characteristics include:

- Plant availability considering refueling intervals and refueling duration
- The collective dose to the plant workers and public in any cases
- The average discharge burnup of fuels

4.16.3 Description of the turbine generator plant systems

One of the most attractive advantages of the SCPR is found in its turbine generator plant systems. Utilizing matured SCP fossil power technologies, the estimated thermal efficiency exceeds 40% with little R&D in the turbine systems except the reheating system.

4.16.3.1 Turbine generator plant

The steam cycle of SCLWR-H employs a two-stage re-heating and eight-stage re-generative system (Fig. 4.16-7), whose thermal efficiency is preliminarily estimated at reaching 42% by Toshiba. The volumetric capacity of the turbines as well as the feedwater heaters used in SCLWR-H is much smaller than those used in conventional LWR plants because of the small volumetric flow rate per electricity production resulted from high enthalpy/pressure of supercritical coolant.

The low-pressure turbine is a four flow, tandem compound, 1500 rpm (for 50 Hz) machine with 41 inches last stage blades. The turbine has one dual-exhaust high-pressure (HP) section, one dual-exhaust intermediate-pressure (IP) section and two dual-exhaust low-pressure (LP) sections. The cycle uses moisture separator reheaters (MSH) with two-stage reheat. This cycle produce about 1000 MWe with a thermal reactor output of 2273 MW.

[1] BWRs have been designed not to operate in hot-standby mode in Japan. There is no occasion to use high-pressure RHR.

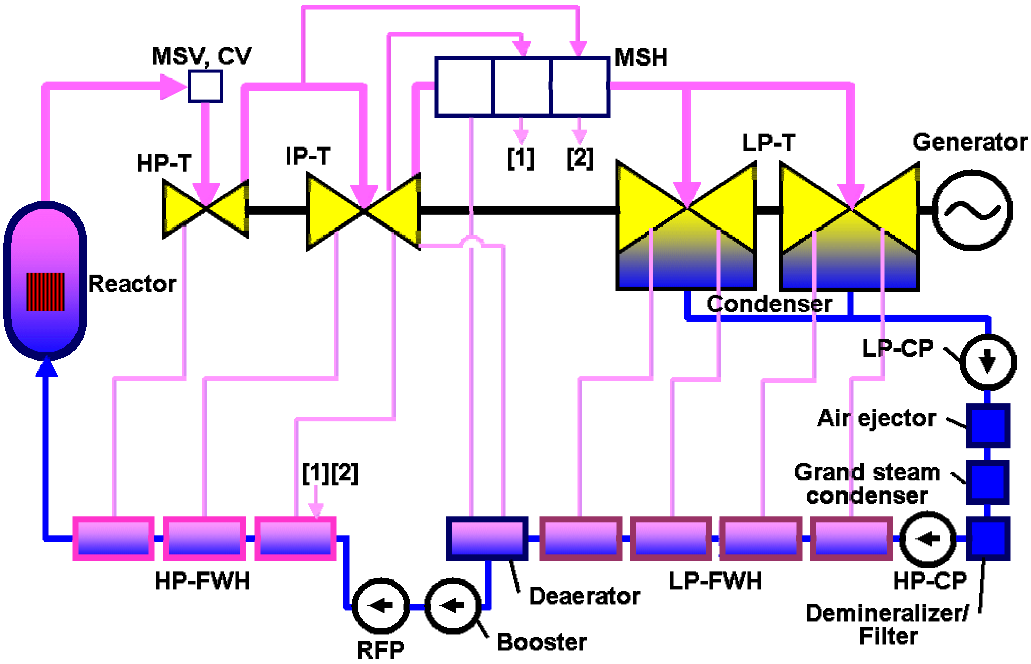


FIG. 4.16-7. SCLWR-H turbine generator plant system proposed by Toshiba

4.16.3.2 Condensate and feedwater systems

Steam bled from the HP, IP and LP sections is conveyed to high (HP-FWH) and low-pressure feedwater heaters (LP-FWH), respectively. The feedwater heaters provide a final feedwater temperature of 280ºC at the rated condition.

The condensate collected in the condensers is pumped by low-pressure condensate pumps (LP-CP) and passes an air ejector, a grand steam condenser and a set of demineralizer and filters. Being pumped by high-pressure condensate pumps (HP-CP), the condensate passes the LP-FWHs including deaerator. Then the condensate is pumped up to supercritical pressure of about 27 MPa and passes HP-FWHs to be heated.

4.16.3.3 Auxiliary systems

Same systems as current LWR or SCP fossil plants utilize are applicable.

4.16.4 Instrumentation and control systems

Instrumentation and control systems for SCPR are being designed based on current BWR and SCP fossil plant technologies.

4.16.5 Electrical systems

Electrical systems for SCPR are being designed based on current BWR and SCP fossil plant technologies. There are few systems unique to SCPR in this area.

4.16.6 Safety concept

The SCPR safety is built on experiences of successful operation of hundreds of LWRs as well as successful operation of number of SCP fossil power plants in the world.

4.16.6.1 Safety requirements and design philosophy

The SCPR design philosophy toward those three Safety and Reliability (SR) goals are following:

- SR-1: Maximum utilization of the matured, proven technologies that have been accumulated in the successful commercial operation of LWRs as well as SCP fossil plants for more than 30 years. Experiences excel theory in safety and reliability of commercial (real) power plants.
- SR-2: Safety system development based on inherent feature of water-cooled reactor and well-developed LWR safety technologies. The inherent feature includes negative void (density) and Doppler coefficients, and relatively large coolant inventory in the RPV as well as in the containment. The well-developed LWR safety technologies mainly include reactivity control systems, coolant makeup systems and core cooling systems. It must be noted that the total core damage frequency (CDF) of ABWR with such safety technologies, which is most recent LWR design in operation, is estimated at far below 10^{-6}. This fact supports that this SR-2 design philosophy.
- SR-3: Utilization of passive technologies for containment and/or RPV protection without sacrificing economics. The principle idea is that few measures are necessary for SR-3 goal if the total CDF is extremely low and this is the case. SCPR is, however, being designed to accommodate passive containment cooling systems (PCCS) because PCCS has been already developed for BWRs and because PCCS will impact little on economics.

4.16.6.2 Safety systems and features (active, passive and inherent)

Safety systems of SCLWR-H mainly consist of a reactor core isolation cooling system (RCIC), two high-pressure auxiliary feedwater systems (AFS), eight automatic depressurization systems (ADS), and three low-pressure core injection systems (LPCI) that also work as RHR (Fig. 4.16-4) [5][6]. Reactor scram, AFS and LPCI are actuated by low core flow rate signals instead of low water level signals as used for current BWRs.

All the systems are designed so that the thermal hydraulic parameters would not exceed the criteria listed in Table 4.16-II in case of a "transient" or an "accident" (Table 4.16-III). The primary criterion for transients is settled from a perspective of the fuel integrity by University of Tokyo that the maximum cladding temperature should not exceed 840°C for nickel-alloy cladding (610°C for stainless steel cladding). This criterion depends on the fuel rod design including the cladding strength. The primary criterion for accidents is settled by the U.S.NRC that the cladding temperature should not exceed 1260°C to avoid fuel failure using stainless steel cladding in LWRs.

TABLE 4.16-II. TRANSIENT AND ACCIDENT CRITERIA FOR SCLWR-H [5][6]

	Transient	Accident
Cladding temperature	< 840°C for Ni-alloy cladding	< 1260°C
Fuel enthalpy	< 272 J/kg	< 963 J/kg
Reactor pressure	< 105% of design pressure	< 110% of design pressure

TABLE 4.16-III. TRANSIENTS AND ACCIDENTS FOR SCLWR-H [5][6]

Transient	Accident
• Loss of feedwater heating	• Loss of coolant
• Inadvertent start-up of AFS	• Total loss of reactor coolant flow
• Partial loss of reactor coolant flow	• Reactor coolant pump trip
• Loss of off-site power	• Control rod ejection under hot standby condition
• Loss of load with turbine bypass	• Control rod ejection under cold standby condition
• Loss of load without turbine bypass	
• Control rod withdrawal in normal operation	
• Control rod withdrawal under hot standby condition	

Potential in safety and reliability has been demonstrated by the University of Tokyo through both deterministic and probabilistic approach [5][6]. In all the transients and accidents shown in Table 4.16-III, the criteria (Table 4.16-II) are not exceeded. A simplified probabilistic safety assessment (PSA) on SCFBR[1] using Japanese PSA data shows that the total CDF is potentially smaller than those of most operating LWRs mainly thanks to its diversified and redundant water injection systems including a turbine-driven AFS, a motor-driven AFS, three LPCIs, and an RCIC.

4.16.6.3 Severe accidents

The PCCS is being incorporated in the SCPR containment to prevent from being over-pressured under severe accident conditions. The PCCS is a passive system which maintains the containment within its design pressure by condensing steam in the containment. It has been developed for SBWR at first then it is being considered to use for some advance nuclear systems including ABWR-II [7].

The PCCS consists of a steam line, a heat exchanger (HX) in a large water pool (PCCS pool) which is placed outside of the containment, a condensate drain and non-condensable gas vent line [8]. Following an accident, high-pressure steam comes from the RPV to the drywell (DW) in the containment and/or steam generated in the DW raises the DW pressure. The resultant pressure difference between the DW and the suppression-chamber (SC) drives mixture of steam and non-condensable gas in the DW into the PCCS-HX through the steam line. The mixture enters the HX in which the steam is condensed by heat exchange between the steam and the water in the PCCS pool. The condensate and non-condensable gas is released to the SC through the drain and vent line. Through these processes, the decay heat is transferred to the PCCS pool and the containment is maintained within its design pressure.

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[1] This is a fast spectrum concept designed by University of Tokyo. The safety systems for SCFBR are identical to those for SCLWR-H [6].

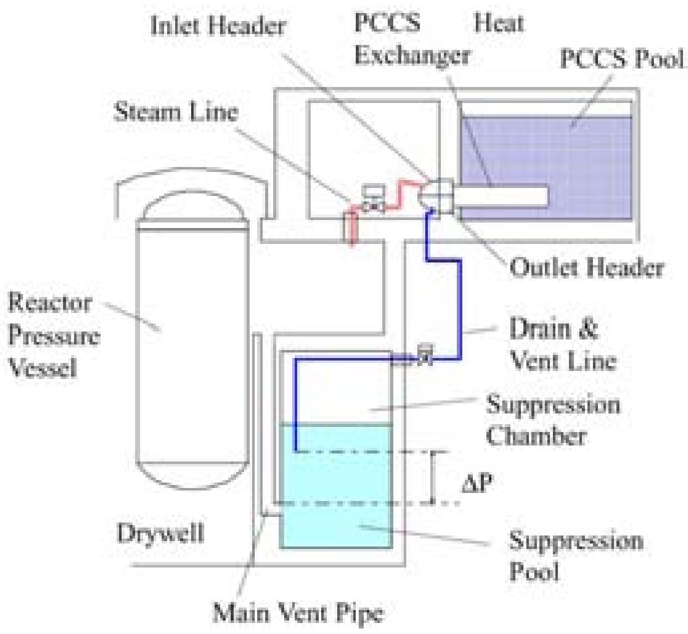


FIG. 4.16-8. PCCS configuration (example) [8]

4.16.7 Plant layout

Plant layout is unnecessarily changed from current ABWR plants [9] except the geometric sizes because the SCPR plant consists of a reactor building accommodating a direct cycle nuclear reactor, a turbine building accommodating a set of steam turbine generator systems and others. The geometric volumes of the reactor building as well as the turbine building will be smaller per electricity generation than those for current LWR plants because of the higher thermal efficiency and simpler/more compact systems.

The SCPR plant will include all buildings which accommodate systems and components related to nuclear system, turbine generator system and control systems for those systems and components. The major buildings are:

- Reactor building
- Service building
- Control building
- Turbine building
- Radwaste building

The SCPR building arrangements is being developed based on the ABWR's criteria [9]:

- Adopt the passive and well established ABWR suppression containment technology.
- Optimize system layout to improve personnel access and equipment maintenance.
- Locate large equipment to facilitate short or modular construction as much as possible.
- Design the reactor building as barrier to post-accident fission product release from the containment, and to external accidents including missiles.
- Align the turbine systems with their axis in-line with the reactor building to minimize the possibility of turbine missile impact.
- Etc.

4.16.8 Technical data (all tentative)

General plant data

Power plant output, gross	1000	MWe
Power plant output, net		MWe
Reactor thermal output	2270	MWth
Power plant efficiency, net	44	%
Cooling water temperature	28.0	ºC

Nuclear steam supply system

Number of coolant loops	1	
Primary circuit volume		m^3
Steam flow rate at nominal conditions	1156	kg/s
Feedwater flow rate at nominal conditions	1156	kg/s

Reactor coolant system

Primary coolant flow rate	1156	kg/s
Reactor operating pressure	25	MPa
Steam temperature/pressure	508/25	°C/MPa
Feedwater temperature	280	°C
Core coolant inlet temperature	280	°C
Core coolant outlet temperature	508	°C
Mean temperature rise across core	228	°C

Reactor core

Active core height	4.2	m
Equivalent core diameter	3.3	m
Heat transfer surface in the core	3889	m^2
Average linear heat rate	18.7	kW/m
Fuel weight	64	tU
Average fuel power density	36	kW/kg U
Average core power density	62	kW/l
Thermal heat flux, F_q	584	kW/m^2
Enthalpy rise, F_H	1965	kJ/kg
Fuel material	Sintered UO_2	
Fuel (assembly) rod total length		mm
Rod array	(25 x 25), square lattice	
Number of fuel assemblies	96	
Number of fuel rods/assembly	301	
Number of spacers		
Enrichment (range) of first core, average		Wt%
Enrichment of reload fuel at equilibrium core	6	Wt%
Operating cycle length (fuel cycle length)		months
Average discharge burnup of fuel [capability]		MWd/t
Cladding tube material	SS304	
Cladding tube wall thickness	0.63	mm
Outer diameter of fuel rods	10.2	mm
Fuel channel/box; material	SS304	
Overall weight of assembly, including box		kg
Uranium weight/assembly	700	kg
Active length of fuel rods	4.2	m
Burnable absorber, strategy/material	axial and radial grading / Gd_2O_3 mixed with fuel	
Number of control rods	96	
Absorber material	B_4C and Hafnium	
Drive mechanism		
Positioning rate		mm/s
Soluble neutron absorber		

Reactor pressure vessel

Inner diameter of cylindrical shell	4260	mm
Wall thickness of cylindrical shell	390	mm
Total height, inside	15000	mm
Base material cylindrical shell	SFVQ1A (ASTM A508)	
RPV head	SFVQ1A (ASTM A508)	
lining	Stainless steel	
Design pressure/temperature	26.5/330	MPa/°C
Transport weight (lower part w/rigging)	708	t
RPV head	192	t

Reactor recirculation pump

Type	None	
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 reinforced concrete	
Overall form (spherical/cyl.)	Cylindrical modified ABWR	
Dimensions (diameter/height)		m
Separated Drywell		m
Design pressure/temperature		kPa/°C
Design leakage rate		vol%/day
Is secondary containment provided?	Yes	

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure (100°C)		kg/s
Coolant injection,	at high pressure (HPCF)		kg/s
	(ARCIC)		kg/s
	at low pressure (LPFL)		kg/s

Power supply systems

Main transformer,	rated voltage		kV
	rated capacity		MVA
Plant transformers,	rated voltage (secondary)		kV
	rated capacity		MVA
Start-up transformer	rated voltage (secondary)		kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)			kV
Number of low voltage busbar systems			
Standby diesel generating units;	number		
	rated power		MW
Standby gas turbine generating units;	number		
	rated power		MW
Number of diesel-backed busbar systems			
Voltage level of these			kV AC
Number of gas turbine-backed busbar systems			
Voltage level of these			kV AC
Number of DC distributions			
Voltage level of these			V DC
Number of battery-backed busbar systems			
Voltage level of these			V DC

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	four flow, tandem compound	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/1 IP/2 LP, double reheat	
Turbine speed	1500	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	24/502	MPa/°C

Generator

Type	3-phase, turbo-generator	
Rated power		MVA
Active power	1000	MW
Voltage		kV
Frequency	50	Hz
Total generator mass, including exciter		t
Overall length of generator		m

Condenser

Type	shell type (2 shells)	
Number of tubes	2 tube bundles/shell	
Heat transfer area		m^2
Cooling water flow rate		m^3/s
Cooling water temperature		^{o}C
Condenser pressure (HP shell)	5	kPa

Condensate pumps

Number		
Flow rate		kg/s
Pump head		MPa
Temperature	34	^{o}C
Pump speed		rpm

Condensate clean-up system

Full flow/part flow

Filter type (deep bed or rod type)

Feedwater tank

Volume		m^3
Pressure/temperature		MPa/^{o}C

Feedwater pumps

Number		
Flow rate	1156	kg/s
Pump head	27.5	MPa
Feed pump power		MW
Feedwater temperature (final)	280	^{o}C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages,	low pressure	4
	high pressure	3
	feedwater tank	1 (deaerator)

4.16.9 Measures to enhance economy and maintainability

4.16.9.1 Design simplification

Thermal components such as heat exchangers and turbines are compact and the need for steam separation systems, re-circulation systems as well as steam generators, is eliminated because of the higher enthalpy content of the Super-critical Water (SCW) and because of no phase change in supercritical regime.

4.16.9.2 Cost reduction of equipment and structures

The high thermal efficiency contributes to reducing the construction cost because the amount of fuel loaded in the reactor pressure vessel (RPV) is smaller per electricity generation. This results in smaller RPV as well as smaller containment vessel and reactor building, for instance. The compact thermal components and elimination of steam separation systems also contribute to reducing the construction cost both in the nuclear reactor island and in the turbine island. Those compact/simple designs improve the operation and maintenance (O&M) cost, too.

4.16.9.3 Reduction of construction period

The utilization of the matured SCP fossil technologies reduces not only the SCPR construction cost but shortens the construction period because those SCP fossil components are well standardized.

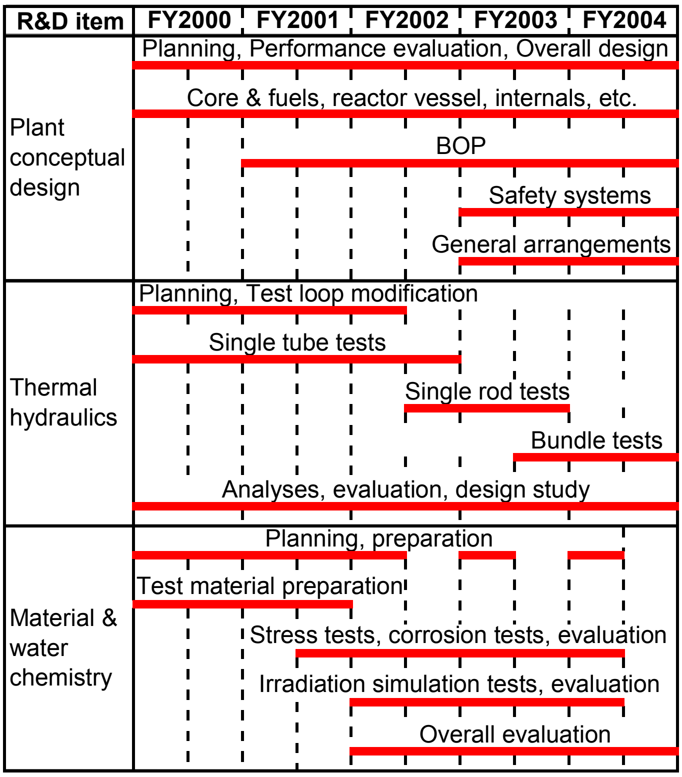
4.16.9.4 Reduction of power generating cost

In addition, fuel cycle cost will be reduced because the estimated thermal efficiency exceeds 40% due to supercritical-pressure (SCP), high-temperature steam at the turbine inlet (Fig. 4.16-1). These features potentially reduces SCPR's power generating cost compared with the conventional LWRs.

4.16.10 Project status and planned schedule

A SCPR development project started in fiscal year (FY) 2000 by a joint team consisting of Univ. of Tokyo, Kyushu Univ., Hokkaido Univ., Hitachi, Ltd, and Toshiba Co. with a national fund in Japan. The objective of this project is to provide technical information essential to SCPR demonstration through concentrating three sub-themes: plant conceptual design, thermal hydraulics, and material & water chemistry. The active duration of this project is four years; the major R&D items as well as schedules are shown in Table 4.16-IV [1].

TABLE 4.16-IV. SCHEDULE OF THE SCPR DEVELOPMENT PROJECT [1]



The major targets of each sub-theme are as follows.

- Plant conceptual design sub-theme; Optimization as well as rationalization as a nuclear system is being performed through design study of key components/systems essential to the SCPR concept. The target is large reduction in amount of volume of such components/systems compared with the conventional LWRs while achieving high thermal efficiency (more than 40 %) without sacrificing the level of safety.
- Thermal hydraulics sub-theme; Heat transfer characteristics of supercritical water as a coolant of the SCPR are examined experimentally and analytically focusing on minimizing the design margin. The experiments are being performed using substitute fluid (Freon-22/R22) for water in a fossil boiler test facility at Kyushu Univ., Japan. The experimental results are being incorporated in LWR analytical tools together with an extended steam/R22 table. These approach will assist in reducing uncertainty in the heat transfer characteristics, so that feasibility of the SCPR will be confirmed.
- Material & water chemistry sub-theme; Promising material candidates for fuel cladding and internals of the SCPR are being chosen from commercially available materials mainly through mechanical test, corrosion test, and simulated irradiation test under the SCPR condition considering water chemistry. In particular, stress corrosion cracking (SCC) susceptibility of the materials is being investigated as well as uniform corrosion and swelling characteristics. Influences of water chemistry on the corrosion product characteristics are also being examined to find suitable water condition as well as to develop rational chemistry controlling methods.

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4.17 RMWR (JAERI, JAPAN)

4.17.1 Introduction

The purposes of the Reduced-Moderation light Water Reactor (RMWR) are to ensure the sustainable energy supply, to meet the flexible demand of plutonium, and to reduce the spent fuel accumulation, being based on the well-experienced LWR technologies. In order to accomplish these purposes, a high conversion ratio beyond 1.0 is an essential requirement from the reactor physics point of view. To obtain such a high conversion ratio in a LWR core, the neutron energy should be increased by reducing the water to fuel volume ratio. The in-core water volume can be reduced by introducing tight-lattice fuel assemblies. Increasing the void fraction also contributes to the reduction of neutron moderation in the core. The resultant neutron spectrum is similar to that in fast breeder reactors (FBRs), and much harder than that in LWRs as shown in Figure 4.17-1.

An effective utilization of uranium resources can be attained by breeding the fissile materials as in the case of FBR. Figure 4.17-2 shows a comparison of cumulative consumption of natural uranium with and without introducing the RMWR. In the analysis shown in Figure 4.17-2, nuclear power capacity in Japan is assumed to be constant at 80GWe after year 2050. When the RMWRs are introduced, natural uranium consumption can be limited to moderate finite levels because the conversion ratio exceeds one.

The high conversion ratio also results in maintaining the plutonium quality, which makes it possible to recycle plutonium many times, i.e. multiple recycling, as shown in Figure 4.17-3. This is a good method for deployment of plutonium and contributes to the reduction of spent fuel accumulation for the long term.

The excess reactivity for the high burn-up and long operation cycle is expected to be reduced due to the high conversion ratio of the RMWR as shown in Figure 4.17-4. The characteristics of high burn-up and long operation cycle are very beneficial to reduce fuel cycle cost as well as reducing the periodical inspection cost.

Concepts of large scale (1,300 MWe class) RMWR have been developed by Japan Atomic Energy Research Institute (JAERI) since 1997 in collaboration with the Japan Atomic Power Company (JAPC), Hitachi, Toshiba and Mitsubishi Heavy Industries. Apart from the large scale RMWR concepts, a small scale (300 MWe class) RMWR with passive safety features is also being developed by JAERI, JAPC, Hitachi and Tokyo Institute of Technology (TIT) under the innovative and viable nuclear energy technology development program (IVNET) sponsored by the Ministry of Economy, Trade and Industries (METI) of Japan since 2000.

The R&D activities including core and system design studies, thermal-hydraulic experiments, reactor physics experiments, and safety analyses are under way to realize a small scale prototype plant in early 2010s. Commercial plants are expected to be introduced by 2020 for replacement of the current LWRs.

Design studies have been performed both for BWR-type and PWR-type cores. In the following sections, only the large scale BWR-type system will be described, as a representative reduced moderation water reactor.

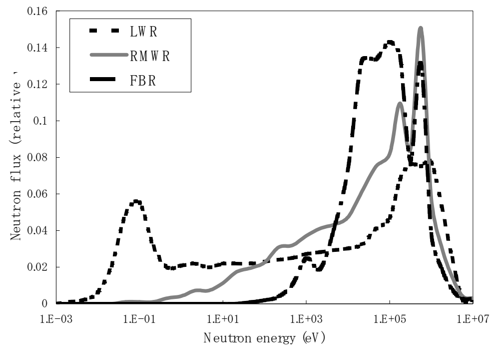


FIG.. 4.17-1. Comparison of neutron spectrum among LWR, RMWR and FBR

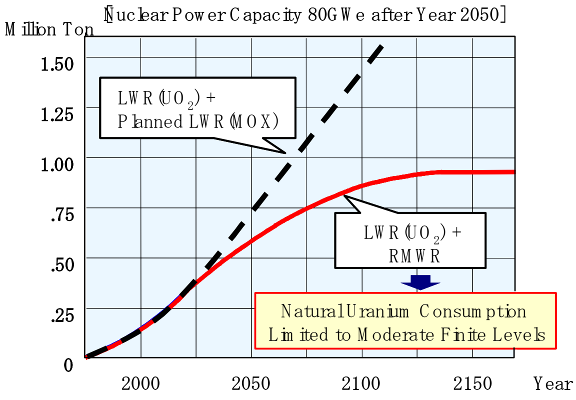


FIG. 4.17-2. Cumulative consumption of natural uranium

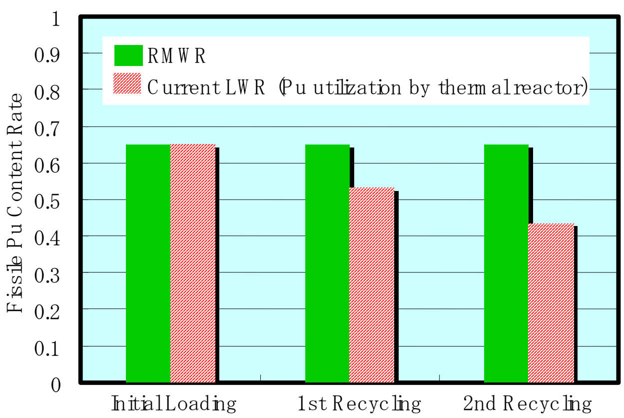


FIG. 4.17-3. Change of the fissile Pu content rate with the recycling

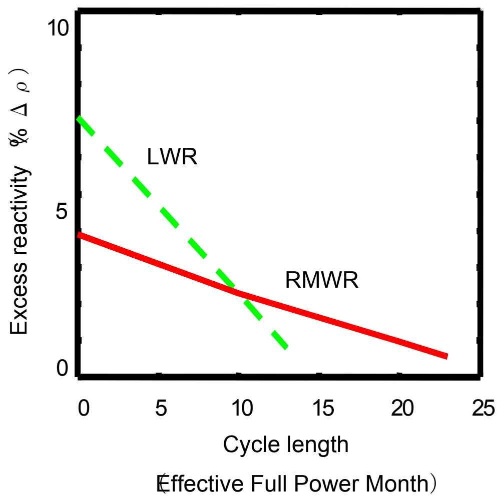


FIG. 4.17-4. Decrease in excess reactivity with burnup

4.17.2 Description of the nuclear systems

4.17.2.1 Primary circuit and its main characteristics

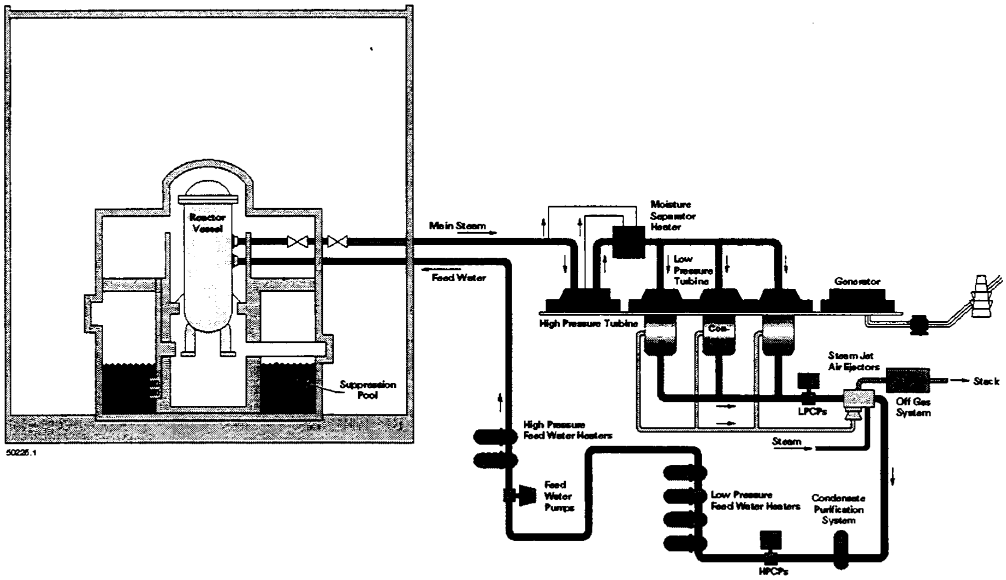


FIG. 4.17-5. ABWR – Steam cycle (see Section 4.1)

Since the RMWR is a BWR-type reactor with the innovative MOX fueled core introduced in the ABWR system framework, the primary circuit and its main characteristics are the same as in the ABWR shown Figure 4.17-5, because the plant system of the RMWR is proposed to be the same as that of the ABWR except for the reactor pressure vessel (RPV) part. The primary functions of the nuclear steam cycle system are:

(1) to deliver steam from the RPV to the turbine main steam system,
(2) to deliver feedwater from the condensate and feedwater system to the RPV,
(3) to provide overpressure protection of the reactor coolant pressure boundary,
(4) to provide automatic depressurization of the RPV in the event of the loss of coolant accident (LOCA) where the RPV does not depressurizes rapidly.

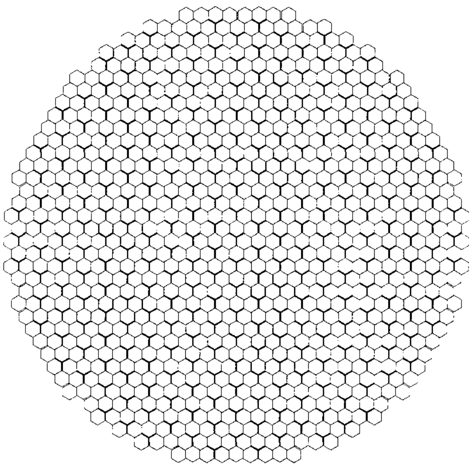
The main steam lines (MSLs) are designed to direct steam from the RPV to the main steam system of the turbine, and the feedwater lines (FWLs) to direct the feedwater from the condensate and feedwater system to the RPV.

There are two main steam isolation valves (MSIVs) welded into each of the four MSLs, one inner MSIV inside the containment and one outer MSIV outside the containment. The nuclear pressure relief system consists of safety/relief valves (SRVs) located on the main steam lines (MSLs) between the RPV and the inboard MSIV. The SRVs are designed to provide three main protection functions: overpressure safety, overpressure relief, and depressurization operation, which is discussed below separately.

The automatic depressurization subsystem (ADS) consists of a part of the SRVs and their associated instrumentation and controls. The ADS designated valves open automatically for events involved with small breaks in the nuclear system process barrier or manually in the power actuated mode when required. The ADS designated valves are capable of operating from either ADS LOCA logic or over-pressure relief logic signals.

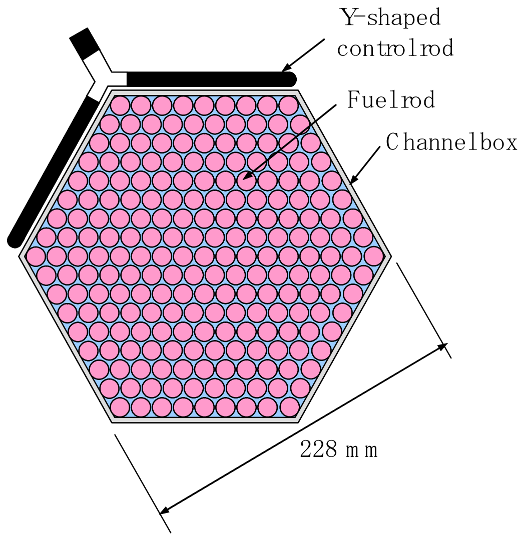
4.17.2.2 Reactor core and fuel design

The RMWR core configuration consists of 900 hexagonal bundles as shown in Figures. 4.17-6 and 4.17-7. The rated core power is 3,926 MWt (1,356 MWe), which corresponds to around 100 kW/l power density. A Y-shaped control rod is adapted in the RMWR core instead of the cross-shaped one used in BWRs, because of better geometrical matching with the hexagonal fuel assembly design. The fuel assembly is in the triangular tight-lattice configuration and contains 217 rods. In order to attain the high conversion ratio, it is necessary to significantly reduce the water to fuel volume ratio in the core. The effective ratio considering the void fraction is less than 0.2 in the present design and about one tenth of that for the ABWR.



Number of fuel assemblies: 900
Number of control rods: 283

FIG. 4.17-6. Core configuration



Number of fuel rods	217
Outer diameter of fuel rods	13.7mm
Fuel rod gap clearance	1.3mm

FIG. 4.17-7. Fuel assembly

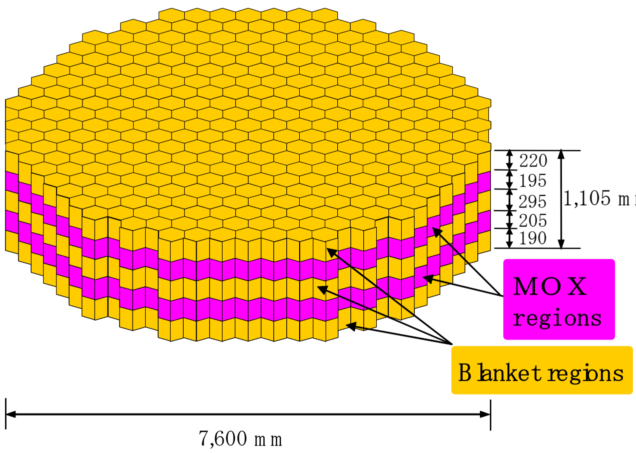


FIG. 4.17-8. Schematic of axial core configuration

Although the ABWR utilizes reactor internal pumps (RIPs) to control the recirculation flow through the core, the present design for the RMWR adopted a natural circulation core cooling system, eliminating RIPs and expanding core region. Another design for 3,188 MWt (1,100 MWe) uses RIPs as in the ABWR. The reactivity control is maintained only by the control rod position, even when RIPs are installed.

The fuel rod is 13.7 mm in diameter and is arranged in the triangular tight-lattice configuration with the gap width of 1.3 mm between rods. There is plutonium content distribution in five groups from 12.0 to 19.3 wt% across the assembly to flatten the local power distribution in the assembly. The average plutonium content in the assembly is 18 wt% and the local power peaking factor is less than 1.05. The axial distribution of MOX in a fuel rod is not homogeneous. There are two MOX regions and three blanket regions of the depleted uranium as shown in Fig. 4.17-8. There is an inner blanket between two MOX regions, and there are the upper and lower blanket regions.

Control rod drive system

The control rod drive (CRD) system is considered to be the same as for the ABWR, *i.e.* the fine motion control rod drive (FMCRD). It is composed of three major elements: the FMCRD mechanisms; the hydraulic control unit (HCU) assemblies, and the control rod drive hydraulic (CRDH) subsystem. The FMCRDs are designed to provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) in response to manual or automatic signals from the reactor protection system (RPS).

However, there are some differences in the control rods from the ABWR design. The control rod is the Y-shaped one as shown in Figure 4.22-7 instead of the cross-shaped one. The control rod material is the enriched boron with the high enrichment of 90 %. And, there exists follower above the control rod made of graphite material to reduce the water region.

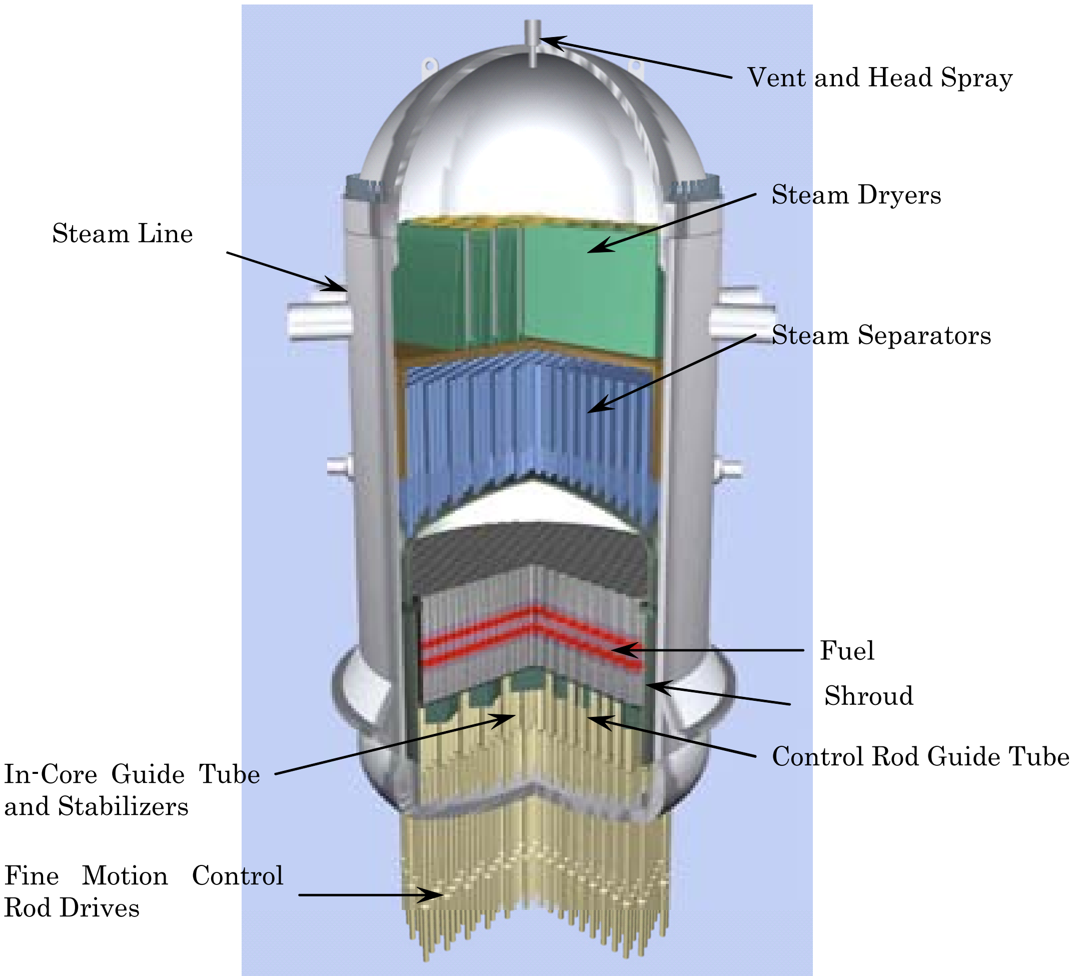


FIG. 4.17-9. Reactor pressure vessel and internals of RMWR

4.17.2.3 Fuel handling and transfer systems

Fuel handling and transfer system are also intended to be the same as in the ABWR system as in Section 4.1.

4.17.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel (RPV) system is basically the same as in the ABWR described in Section 4.1 except for the large diameter for the flat core of RMWR.

The RPV consists of:

(1) the RPV and its appurtenances, supports and insulation, excluding the loose parts monitoring system, and
(2) the reactor internal components enclosed by the vessel, excluding the core (fuel assemblies, control rods, in-core nuclear instrumentation and neutron sources) and control rod drives (CRDs). The RPV system is located in the primary containment. The reactor coolant pressure boundary (RCPB) portion of the RPV and its appurtenances act as a radioactive material barrier during plant operation.

Reactor internals

The RMWR RPV and internals are basically the same as in the ABWR, which are illustrated in Figure 4.17-9 for the case with RIPs. The major reactor internal components in the RPV System are: (1) Core support structures, and (2) Other reactor internals.

The core support structures encompass: the shroud, shroud support and a portion of CRD housings inside the reactor internals RPV, core plate, top guide, fuel supports, and control rod guide tubes (CRGTs).

Other reactor internals are:

- Feedwater spargers, shutdown cooling (SDC) and low pressure core flooder (LPFL) spargers for the residual heat removal (RHR) system, high pressure core flooder (HPCF) spargers and couplings, and a portion of the in-core housings inside the RPV and in-core guide tubes (ICGTs) with stabilizers.
- Surveillance specimen holders, shroud head and steam separators assembly and the steam dryer assembly.

Reactor recirculation pumps

In the representative design with 1,356 MWe power output, the natural circulation core cooling is adopted. However, in a variation of design, the RIPs are adopted as in the ABWR.

4.17.2.5 Reactor auxiliary systems

The auxiliary is basically the same as in the ABWR described in Section 4.1. The main auxiliary systems in the nuclear island consist of the reactor building cooling water (RBCW) system, the reactor water cleanup (RWCU) system, the fuel pool cooling and cleanup (FPCU) system and the suppression pool cleanup (SPCU) system. In addition there are many other auxiliary systems such as instrument and service air, condensate and demineralized water transfer, chilled water, HVAC, equipment drain, floor drain and other systems.

4.17.2.6 Operating characteristics

The RMWR design incorporates the extensive automation of the operator actions which are required during a normal plant startup, shutdown and power range maneuvers. It is basically similar to the ABWR design. The automation features adopted are designed for enhanced operability and improved capacity factor, relative to conventional BWR designs. However, the extent of automation implemented has been carefully selected as in the ABWR to ensure that the primary control of plant operations remains with the operators. The operators remain fully cognizant of the plant status and can intervene in the operation at any time, if necessary.

The control room design is also the same as for the ABWR.

The FMCRDs are moved electronically in small increments during normal operation, allowing precise power management. The FMCRDS are inserted into the core hydraulically during emergency shutdown, with the backup provision for continuous electronic insertion.

4.17.3 Description of turbine generator plant systems

4.17.3.1 Turbine generator plant

The turbine generator plant design is also the same as for the ABWR.

The main turbine

The main turbine is a six flow, tandem compound, single reheat machine. The turbine has one duel-exhaust high-pressure section and three dual-exhaust low-pressure sections. The cycle uses conventional moisture separator reheaters with single stage reheat for the cross-around steam.

Extraction steam from the high and low-pressure turbine extraction nozzles is conveyed to the high and low-pressure feedwater heaters, respectively. The feedwater heating systems are designed to provide a final feedwater temperature at 100 percent nuclear boiling rate.

Turbine bypass system

The turbine bypass system (TBP) provides the capability to discharge main steam from the reactor directly to the condenser to minimize step load reduction transient effects on the reactor coolant system. The TBP is also used to discharge main steam during reactor hot standby and cool-down operations.

The turbine bypass valves are opened by redundant signals received from the steam bypass and pressure control system whenever the actual steam pressure exceeds the preset steam pressure by a small margin. This occurs when the amount of steam generated by the reactor cannot be entirely used by the turbine. This bypass demand signal causes fluid pressure to be applied to the operating cylinder, which opens the first of the individual valves. As the bypass demand increases, additional bypass valves are opened, dumping the steam to the condenser. The bypass valves are equipped with fast acting servo valves to allow rapid opening of bypass valves upon turbine trip or generator load rejection.

Main condenser

The main condenser, which does not serve or support any safety function and has no safety design basis, is a multipressure three-shell type deaerating type condenser. During plant operation, steam expanding through the low-pressure turbines is directed downward into the main condenser and condensed. The main condenser also serves as a heat sink for the turbine bypass system, emergency and high level feedwater heater and drain tank dumps, and various other startup drains and relief valve discharges.

4.17.3.2 Condensate and feedwater systems

The condensate and feedwater system are designed to provide a dependable supply of high- quality feedwater to the reactor at the required flow, pressure, and temperature. The condensate pumps take the deaerated condensate from the main condenser hotwell and deliver it through the steam jet air ejector condenser, the gland steam condenser, the off-gas condenser, the condensate demineralizer, and through three parallel strings of four low pressure feedwater heaters to the reactor feed pumps section. The reactor feed pumps each discharge through two stages of high-pressure heaters (two parallel strings) to the reactor. Each reactor feedwater pump is driven by an adjustable speed synchronous motor. The drains from the high-pressure heaters are pumped backward to the suction of the feed pumps.

Two feedwater lines transport feedwater from the feedwater pipes in the steam tunnel through CV penetrations to horizontal headers. Isolation check valves are installed upstream and downstream of the CV penetrations and manual maintenance gate valve are installed upstream of the horizontal headers.

4.17.3.3 Auxiliary systems

The turbine building cooling water system (TBCW), which is a non-safety related system, removes heat from the auxiliary equipment in the turbine building and transfers this heat to the turbine building service water (TBSW) system. The TBSW system transfers the heat taken from the TBCW system to the power cycle heat sink which is part of the irculating water system.

4.17.4 Instrumentation and control systems

4.17.4.1 Design concept, including control room

The RMWR system design is based on the well-developed current LWR technologies. The system developed for ABWR can be applicable to that of the RMWR. The instrumentation and control system, including the reactor protection system, can also be applicable, so that the similar configuration to that of the ABWR is to be adopted. In the system, the digital controller, interfacing with plant equipment and sensors required for plant operation are to be implemented.

Features of the instrumentation and control system of the ABWR will be reflected in the RMWR design. These features are followings:

- The main control room panels consist of an integrated set of operator interface panels, which are based on the enhanced man-machine interface design.
- Multiplexed signal transmission using fiber optic data links are utilized which significantly reduces the quantities of control cables.
- The safety system logic and controls (SSLC) are performed by microprocessor-based software.

4.17.4.2 Reactor protection and other safety systems

The reactor protection system (RPS) is an overall complex of instrument channels, trip logic, trip actuators, manual controls, and scram logic circuitry that initiates the insertion of control rods to scram the reactor when unsafe conditions are detected. The RPS uses the function of SSLC to perform its functions.

The feedwater control (FWC) system controls the flow of feedwater into the RPV to maintain the water level in the vessel within predetermined limits during all plant operating modes. The feedwater temperature is different from that for ABWR due to the difference of recirculation flow ratio.

One of the essential components required for instrumentation is the neutron monitoring system. This system is to be composed from the startup range neutron monitoring (SRNM) subsystem, the power range neutron monitoring (PRNM) subsystem, the automatic traversing in-core probe (ATIP) subsystem and the multi-channel rod block monitoring (MRBM) subsystem same as the ABWR.

4.17.5 Electrical systems

The electrical system design is also the same as for the ABWR.

4.17.5.1 Operational power supply systems

On-site power is supplied from either the plant turbine generator, utility power grid, or an off-site power source depending on the plant operating status. During normal operation, plant loads are supplied from the main generator through the unit auxiliary transformers. A generator breaker allows the unit auxiliary transformers to stay connected to the grid to supply loads by backfeeding from the switchyard when the turbine is not online.

Direct current power supply

The DC power supply system (DC) consists of three separate subsystems. The system begins at the source terminals of the plant safety and non-safety battery chargers. It ends at the input terminals of the plant DC loads (motor, control loads, etc.) and at the input terminals of the inverters of the low voltage vital AC power supply system.

Instrument and control power supply

The instrument and control power supply system (ICP) provides 120 V AC power to instrument and control loads which do not require continuity of power during a loss of preferred power.

4.17.5.2 Safety-related systems

Class 1E AC power supply

The class 1E buses of the on-site power system consists of three independent divisions of class 1E equipment. Each division is fed by an independent class 1E bus at the medium voltage level, and each division has access to one on-site and two off-site (normal and alternate preferred) power sources.

Each division has access to an additional power source which is provided by the combustion turbine generator (CTG).

Each division is provided with an on-site safety related standby diesel generator which supplies a separate on-site source of power for its division when normal or alternate preferred power is not available. The standby diesel generators are capable of providing the required power to safely shut down the reactor after loss of preferred power and/or loss of coolant accident and to maintain the safe shutdown condition and operate the class lE auxiliaries necessary for plant safety after shutdown.

The on-site standby AC power supplies (diesel generators) have sufficient capacity to provide power to all their respective loads. Loss of the preferred power supply, as detected by undervoltage relays in each division, will cause the standby power supplies to start and automatically connect, in sufficient time to safely shut down the reactor or limit the consequences of a design basis accident (DBA) to acceptable limits and maintain the reactor in a safe condition.

Direct current power supply

The class 1E 125 V DC subsystem consists of four independent and redundant divisions (I, II, III, and IV). All four divisional batteries are sized to supply 125 V DC power to their loads during a design basis accident, coincident with loss of AC power. This sizing of the division I battery also meets the requirement to permit operation of the station blackout coping systems. The division I battery is sized to support operation of RCIC and remote shutdown system (RSD), as well as a minimum necessary emergency lighting. This manual load shedding takes credit for the RCIC operation from outside the main control room.

Vital (uninterruptable) power supply

The class 1E vital AC (VAC) power supply provides redundant, reliable power to the safety logic and control functions during normal, upset and accident conditions. The VAC is comprised of three independent subsystems. Each subsystem supplies uninterruptable, regulated AC power to those loads which require continuity of power during a loss of preferred power (LOPP).

4.17.6 Safety concept

4.17.6.1 Safety requirements and design philosophy

The development of the RMWR reactor systems ranges from those for the near-term deployment to those for the long-term deployment. The RMWR for the long-term deployment systems utilize advanced technologies such as fully passive safety systems and in-vessel steam generator. In this report, only the RMWR for the near-term deployment system will be described.

The development of RMWR aims at the deployment by 2020. Considering this short schedule, it is preferable that the design does not require extensive R&D efforts and significant changes of the current safety regulations in Japan. The safety design for RMWR is, therefore, based on well-matured technologies accumulated for the current generation LWRs especially for ABWR. Although the use of the tight-lattice core with very narrow flow gap makes the core cooling difficult, the disadvantage is compensated by the reduction of the core pressure drop by the short core.

Another difference between ABWR and RMWR is the use of a passive containment cooling system for the accident management measures, for which a large-scale confirmatory test is underway at JAERI.

4.17.6.2 Safety systems and features (active, passive and inherent)

The reactor internal pumps installed in the RPV eliminate recirculation piping outside the RPV. The ECCS is a three-division system, with a high and low pressure injection pump and heat removal capability in each division functioning independently. One of the systems serves as the reactor core isolation cooling (RCIC) system, which has a steam driven high pressure pump. The ECCS adopts three on-site emergency diesel-generators to support core cooling and heat removal if off-site power is lost. The ECCS is designed to maintain core coverage for any postulated line break size during accidents.

The ABWR has the FMCRD, which adopt a diversified control rod drive mechanism with a fine-tunable electric drive system in addition to the current hydraulic drive system. Response to anticipated transients without scram (ATWS) was improved by the FMCRD, which allow reactor shutdown either by hydraulic or electric insertion.

A Reinforced Concrete Containment Vessel (RCCV) is used for the reactor containment for the ABWR. Employment of the natural circulation core cooling or RIPs eliminates piping for the reactor coolant recirculating system. This enables the center of gravity of the reactor building to be lowered. Moreover, the flat core of the RMWR makes the building more resistant to earthquakes than the current ABWR.

4.17.6.3 Severe accidents (beyond design basis accidents)

For the mitigation of the effects of the severe accidents, several accident management (AM) measures are planned as done for ABWR. In addition to those AM measures, a passive containment cooling system (PCCS) is utilized to prevent the containment damage caused by the over-pressurization due to the steam generation during LOCAs. The PCCS is a passive cooling system without relying on the pump operation. It is designed to have sufficient cooling capability for steam condensation with the conservatively estimated amount of noncondensables (nitrogen and/or hydrogen).

The PCCS heat exchanger (HEX) is submerged in the water pool located outside the containment, and is connected to the drywell for the inlet side and the suppression chamber for the outlet side. This PCCS is characterized by the use of horizontal tubes for the HEX, compared to those for the SBWR with vertical tubes. The use of the horizontal HEX has several advantages over the vertical one, which includes the enhancement of the earthquake resistance, the reduction of the pool water level, the easiness of the maintenance. Among the other, the horizontal HEX can be economically optimized, while the optimization is impossible for the vertical HEX because the tube length is limited by the pool liquid level. Large-scale tests are now being conducted at JAERI to confirm the effectiveness of the horizontal PCCS, from which the promising results have been obtained.

4.17.7 Plant layout

The plant layout is also the same as for the ABWR.

4.17.7.1 Buildings and structures, including plot plan

The RMWR plant includes all buildings which are dedicated exclusively or primarily to housing systems and the equipment related to the nuclear system or controls access to this equipment and systems. There are five such buildings within the scope:

(a) Reactor building - includes the reactor pressure vessel, containment, and major portions of the nuclear steam supply system, refuelling area, diesel generators, essential power, non-essential power, emergency core cooling systems, IWAC and supporting systems;
(b) Service building - personnel facilities, security offices, and health physics station;
(c) Control building - includes the control room, the computer facility, reactor building component cooling water system and the control room HVAC system;
(d) Turbine building - houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building;
(e) Radwaste building - houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

The site plan of the RMWR includes the reactor, service, control, turbine, radwaste and supporting buildings.

Development of the RMWR plant and building arrangements has been guided by the following criteria:

(a) Retain the passive and well established pressure suppression containment technology;
(b) Emphasize optimal layout of systems to improve personnel access and equipment maintenance activities.

4.17.7.2 Reactor building

The RMWR integrated reactor building and containment structure has been analyzed for a safe shutdown earthquake of 0.3g.

Key distinguishing features of the RMWR reactor building design include:

(a) Elimination of external recirculation loops reduces the containment volume associated with high construction costs.
(b) Reduced building volume reduces material costs and construction schedule.
(c) Designed with simple structural shapes to improve constructability to reduce capital costs and the construction schedule.
(d) Improved personnel and equipment access for enhanced operability and maintainability.

4.17.7.3 Containment

The RMWR pressure suppression primary containment system is designed to have the following functional capabilities:

(a) The containment structure is designed to maintain its functional integrity during and following the peak transient pressures and temperatures which would occur following any postulated loss of coolant accident (LOCA). A design basis accident (DBA) is defined as the worst LOCA pipe break (which leads to maximum containment pressure and/or temperature), and is further postulated to occur simultaneously with a loss of off site power (LOOP) and a safe shutdown earthquake (SSE).
The containment structure is designed for the full range of loading conditions consistent with normal plant operating and accident conditions including the LOCA related design loads.

(b) The containment structure and isolation, with concurrent operation of other accident mitigation systems, is designed to limit fission product leakage during and following the postulated design basis accident (DBA) to values less than leakage rates which would result in off-site radiation doses greater than those set forth in 10 CFR 100.

(c) Capability for rapid closure or isolation of all pipes or ducts which penetrate the containment boundary is provided to maintain leakage within acceptable limits.

(d) The containment structure is designed to withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.

(e) The containment structure is designed to accommodate flooding to a sufficient depth above the active fuel to permit safe removal of the fuel assemblies from the reactor core after the postulated DBA.

(f) The containment structure is protected from or designed to withstand hypothetical missiles from internal sources and uncontrolled motion of broken pipes which could endanger the integrity of the containment.

(g) The containment structure is designed to provide means to channel the flow from postulated pipe ruptures.

(h) The containment system is designed to allow for periodic tests at the calculated peak or reduced test pressure to measure the leakage from individual penetrations and isolation valves, and the integrated leakage rate from the structure to confirm the leaktight integrity of the containment.

(i) The atmospheric control system (ACS) establishes and maintains the containment atmosphere to less than 3.5% (by volume) oxygen during normal operating conditions. To assure an inert atmosphere, operation of two permanently installed recombiners can be initiated on high levels as determined by the containment atmospheric monitoring system (CAMS).

4.17.7.4 Turbine building

The turbine building houses all the components of the power conversion system. This includes the turbine-generator, main condenser, air ejector, steam packing exhauster, off-gas condenser, main steam system, turbine bypass system, condensate demineralizers, and the condensate and feedwater pumping and heating equipment. The small size of the RMWR turbine building makes a significant contribution to capital cost savings and a shorter construction schedule.

4.17.7.5 Other buildings

No information provided.

4.17.8 Technical data (for the RMWR)

General plant data

Power plant output, gross	1356	MWe
Power plant output, net	1300	MWe
Reactor thermal output	3926	MWth
Power plant efficiency, net	33.1	%
Cooling water temperature	≈ 28	^{o}C

Nuclear steam Supply system

Number of coolant loops	1	
Primary circuit volume		m^3
Steam flow rate at nominal conditions	2122	kg/s
Feedwater flow rate at nominal conditions	2118	kg/s

Reactor coolant system

Primary coolant flow rate	4722	kg/s
Reactor operating pressure	7.2	MPa
Steam temperature/pressure	288 / 7.2	^{o}C/MPa
Feedwater temperature	277	^{o}C
Core coolant inlet temperature	283	^{o}C
Core coolant outlet temperature	288	^{o}C
Mean temperature rise across core	5	^{o}C

Reactor core

Active core height	1.105	m
Core diameter	7.60	m
Heat transfer surface in the core	9288	m^2
Average linear heat rate	18.19	kW/m
Fuel weight		tU
Average fuel power density		kW/kg U
Average core power density		kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		
Fuel material	UO_2-PuO_2 / UO_2	
Fuel (assembly) rod total length	1355	mm
Rod array	triangular lattice	
Number of bet assemblies	900	
Number of fuel rods/assembly	217	
Number of spacers	5	
Enrichment (range) of first core, average		Wt%
Enrichment of reload fuel at equilibrium core	18(Ave. Pu cont.)	Wt%
Operating cycle length (fuel cycle length)	24	months
Average discharge burnup of fuel [capability]	45,000	MWd/t
Cladding tube material	Advanced SS	
Cladding tube wall thickness	0.6	mm
Outer diameter of fuel rods	13.7	mm
Fuel channel/box; material	SS	
Overall weight of assembly, including box		kg
Uranium weight/assembly		kg
Active length of fuel rods	1105	mm
Burnable absorber, strategy/material		
Number of control rods	283	
Absorber material	B_4C	
Drive mechanism	Fine Motion Electric Motor/Hydraulic	
Positioning rate		mm/s
Soluble neutron absorber		

Reactor pressure vessel

Inner diameter of cylindrical shell	8900	mm
Wall thickness of cylindrical shell	170	mm
Total height, inside	19400	mm
Base material : cylindrical shell		
RPV head		
Lining	stainless steel	
Design pressure/temperature	8.62 / 302	MPa/ ^{o}C
Transport weigh (lower part w/rigging)		t
RPV head		t

Primary containment

Type	Pressure-suppression/ reinforced concrete	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	29 / 36	m
Design pressure/temperature	310 / 171	kPa/ ^{o}C
Design leakage rate	0.4	vol%/day
Is secondary containment Provided?	Yes	

Reactor auxiliary systems

Reactor water cleanup,	capacity	42.36	kg/s
	filter type	deep bed	
Residual heat removal,	at high pressure		kg/s
	at low pressure (100 ^{o}C)	253.8	kg/s

Coolant injection,	at high pressure (HPCF)	36.3	kg/s
	at low pressure (LPCF)	253.8	kg/s

<u>Power supply systems</u>

Main transformer,	rated voltage		kV
	rated capacity		MVA
Plant transformers,	rated voltage		kV
	rated capacity		MVA
Start-up transformer	rated voltage		kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)			kV
Number of low voltage busbar systems			
Standby diesel generating units: number		3	
rated power		6.57	MW
Number of diesel-backed busbar systems		3	
Voltage level of these		4160	V AC
Number of DC distributions			
Voltage level of these			V DC
Number of battery-backed busbar systems			
Voltage Level of these			V AC

<u>Turbine plant</u>

Number of turbines per reactor	1	
Type of turbine(s)	six flow, tandem compound, single reheat	
Number of turbine sections per unit (e.g. HP/LP/LP)		
Turbine speed	1500	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	6.79 / 284	MPa/ °C

<u>Generator</u>

Type	3-phase, turbo-generator	
Rated power	1540	MVA
Active power	1356	MW
Voltage	27	kV
Frequency	50	Hz
Total generator mass, including exciter		t
Overall length of generator		m

<u>Condenser</u>

Type	shell type	
Number of tubes	1 tube pass/shell	
Heat transfer area	124,170	m^2
Cooling water flow rate	34.68	m^3/s
Cooling water temperature		°C
Condenser pressure (HP shell)	11.75	kPa

<u>Condensate pumps</u>

Number	4	
Flow rate	≈ 435	kg/s
Pump head	3.82	MPa
Temperature		°C
Pump speed		rpm

<u>Condensate clean-up system</u>

Full flow/part flow	Full flow
Filter type (deep bed or rod type)	deep bed

<u>Feedwater tank</u>

Volume		m^3
Pressure/temperature		MPa/ °C

<u>Feedwater pumps</u>

Number	3	
Flow rate	≈ 1000	kg/s
Pump head	6	MPa
Feed pump power		MW
Feedwater temperature (final)	277	°C
Pump speed		rpm

<u>Condensate and feedwater heaters</u>

Number of heating stages,	low pressure	3 x 4
	high pressure	2 x 2
	feedwater tank	

4.17.9 Measures to enhance economy and maintainability

Since the RMWR is a BWR-type reactor with the innovative MOX fueled core introduced in the ABWR system framework, the main design measures aimed to improve the plant economics are the same as the existing ABWR or the future designs (Ref. 1-3, see also Section 4.1), because the plant system of the RMWR is proposed to be the same as that of the ABWR except for the reactor pressure vessel (RPV) part. Those features are advanced-type control-rod drive mechanism (FMCRD), integrated digital instrumentation and control system, large capacity and high efficiency turbine system and so on. Operation experiences accumulated in Kashiwazaki-Kariwa Nuclear Power Station units No.6 & 7 are also beneficial.

In the reference design of RMWR, the difference from the ABWR plant is that the natural circulation core cooling system is adopted and the reactor internal pumps are not installed. That can be possible due to relatively small pressure loss through the core presently designed. The elimination of RIP simplifies the design and reduces the cost.

Since the RMWR aims at multiple recycling of plutonium for the long-term energy supply with the uranium resources, reduction of the fuel cycle cost is important. There are two measures for it. One is to increase the burn-up as much as possible. The other is to reduce the reprocessing cost. A simplified PUREX type reprocessing process has been proposed by JAERI, eliminating the purification processes for uranium and plutonium after their separation process, but keeping the decontamination factor of about 10^5 and reduce the reprocessing cost approximately one half.

4.17.10 Project status and planned schedule

Concepts of large scale (1,300 MWe class) RMWR have been developed by Japan Atomic Energy Research Institute (JAERI) since 1997 in collaboration with the Japan Atomic Power Company (JAPC) and Japanese vendors (Hitachi, Toshiba and Mitsubishi Heavy Industries). Apart from the large scale RMWR concepts, a small scale (300 MWe class) RMWR with passive safety features is also being developed by JAERI, JAPC, Hitachi and Tokyo Institute of Technology (TIT) under the innovative and viable nuclear energy technology development program (IVNET) sponsored by the Ministry of Economy, Trade and Industries (METI) of Japan since FY2000.

The R&D activities including core and system design studies, thermal-hydraulic experiments, reactor physics experiments, and safety analyses are under way to realize a small-scale prototype plant in early 2010s. Commercial plants are expected to be introduced by 2020 for replacement of the current LWRs.

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4.18 RBWR (HITACHI LTD., JAPAN)

4.18.1 Introduction

An adaptation of the BWR has been conceived which demonstrates a positive feature of the BWR - specifically the potential to breed and to consume multi-recycled plutonium. The concept is referred to as the Resource-Renewable Boiling Water Reactor (RBWR) [Ref. 1]. The RBWR may be characterized as a BWR which operates with mixed (depleted uranium and plutonium) oxide fuel with an epi-thermal spectrum, and which has a breeding ratio of 1.0 and a negative void coefficient. The RBWR uses a short core (less than 1.5m active core length). This shorter nuclear core should have a desirable effect upon nuclear island economics and capital equipment cost reductions. The RBWR is also effective in ensuring actinide transmutation and conductive to nuclear non-proliferation.

With a breeding ratio of approximately 1.0, this system, operating symbiotically with LWRs of current design, has the potential to serve as a long term energy supply greatly increase the energy that can be obtained from the uranium resource base. The RBWR is thus a recycle oriented BWR utilizing uranium and plutonium discharged from LWRs.

In the second phase of the RBWR development, it is foreseen that other actinide elements as well as plutonium would be recycled together thereby playing an important role from the viewpoint of storage of long-life actinide elements.

The quantity of fissile plutonium in a spent fuel bundle of the RBWR with a breeding ratio of 1.0 is nearly equal to that in a fresh bundle. Furthermore, the RBWR can be operated with more than 35% plutonium-240 content under the equilibrium condition of plutonium multi-recycling. If international nuclear fuel centers comprised of fabrication facilities and reprocessing facilities were established, domestic and international transfers of plutonium could be reduced or avoided, except for the transfer of MOX fuel bundles between a center and the RBWRs. This approach may offer nuclear non-proliferation advantages (see the Section on proliferation resistance of nuclear power in the Introduction to this TECDOC).

4.18.2 Description of the nuclear systems

4.18.2.1 Primary circuit and its main characteristics

The first design target is to realize long-term energy supply under the condition that power generation cost and safety are at ABWR [Ref. 2] levels. Other design goals, including recycle of actinides other than plutonium will be addressed later in the design effort.

The core design includes a hexagonal tight lattice fuel bundle, instead of the square lattice of current BWRs, a coolant with high void fraction and Y-type control rods instead of cruciform control rods can reduce the effective water-to-fuel volume ratio to about 0.3. Mixed oxide fuel, consisting of plutonium oxide with depleted uranium oxide, with an effective water-to-fuel volume ratio of less than 0.3 is utilized to provide a breeding ratio of more than 1.0. The tight lattice fuel bundles are shorter than the ABWR bundles. The same rated power, burnup, reactor pressure vessel diameter and cladding material as those of ABWR are utilized with the goal of maintaining plant construction cost for the RBWR at the ABWR level. Depleted uranium, rather than natural uranium, is used in the MOX fuel to reduce the fuel cost. Enhancement of neutron leakage in the axial direction by the shortened core and the swing of the power distribution by axially zoned or axially heterogeneous fuel bundles make the void coefficient negative. Holding the core exit quality under 40% is essential to retain radioactive corrosion products and radioactive materials generated from the fuel rod in the case of fuel failures in the pressure vessel.

4.18.2.2 Reactor core and fuel design

Main design parameters of the RBWR in the equilibrium core and specifications of fuel bundle are compared with those of an ABWR in reference 1, which describes different core design options for the RBWR, including: the RBWR-HO which is a homogeneous core using axially two-zoned fuel bundles minimizing plutonium inventory; the RBWR-B2 which is a high burnup core with average exposure of discharged fuel bundle in the equilibrium core of 70GWd/t; and the RBWR-AC which is a core for recycle of minor actinides with plutonium.

All of these reactor design options have 1350MWe rated electric power, 7.2MPa core pressure and 2.88m core radius. The core and fuel design is illustrated in Figure 4.18-1. The 223 Y-type control rods with three wings are inserted into gaps among the 720 hexagonal fuel bundles to constitute the nuclear core. Followers consisting of zircaloy are attached to control rods to prevent water from going into spaces after rods are withdrawn. The triangle fuel rod lattice is effective in decreasing the ratio of water to fuel while keeping cooling ability. The core height of the RBWR-B2 option is 70cm to which the axial blankets, 25cm for the upper one and 20cm for the lower one, are attached. The core coolant flow is 3.2×10^4 tons/h with sub-cooling of 10℃ at the entrance and 28w/o steam quality at the core exit which keeps the distillation function, while confining radioactive corrosion products to the pressure vessel, even in a transient process.

Lattice pitches of fuel bundles are 199.3mm for the side with the control rod and 194.4mm for the side without one. The channel box of the fuel bundle is hexagonal with 193.6mm length for facing sides and 2mm thickness. Control rod wings are 7.5mm thick and the gap between a wing and the channel box of a fuel bundle is 1.6mm on each side. Fuel bundles for the side without control rods make contact through pads on the channel box and there is the space 0.8mm wide between channel boxes. The 271 fuel rods are arranged in a triangular lattice of 11.38mm pitch and 1.3mm space to constitute the fuel bundle which has fuel rods fixed with five ring-type spacers keeping 1.05mm gap between the outermost fuel rod and the channel box. Fuel rods for the equilibrium core of the RBWR-B2 option contain a mixture of PuO_2 and UO_2 and have a 10.08mm outside diameter. Bundle-averaged fissile plutonium enrichment of 18w/o in the upper core of 22cm height and in the lower core of 19cm height is required and internal blanket of 29cm height consisting of depleted uranium is arranged between the upper core and lower one. In the RBWR-B2, 10 tons per 1GWe of internal and external fissile plutonium inventory is required.

4.18.2.3 Fuel handling and transfer systems

Fuel handling and transfer systems are basically the same as those of ABWR. Because of the short core, spent fuel pool that locates above the RPV is shallower than ABWR, which leads plant construction cost reduction.

A breeding ratio of 1.01 is achievable with an exposure of 70GWd/t. The internal breeding ratio, the number of fissile plutonium atoms left in the core part of discharged fuel bundles per atoms in initial fuel bundles, is 0.90. This large internal breeding ratio results in a small net percentage fissionable material burnup in the core and, therefore, a small reactivity swing between refuelings. Loading pattern of fuel bundles in the equilibrium core adopts the zone loading and periodic boundary condition of 120 degrees in the azimuthal direction due to flattening of the radial power peaking to 1.20. The core coolant flows in the lower blanket with 10℃ sub-cooling and gives a void fraction of about 30% at the bottom of the core due to heating in the lower blanket and that of about 80% at the top of the core. Maximum linear heat generation rate of 13.1kW/ft and minimum critical power ratio (MCPR) of 1.40 in Haling power distribution [Ref. 3] can be achieved.

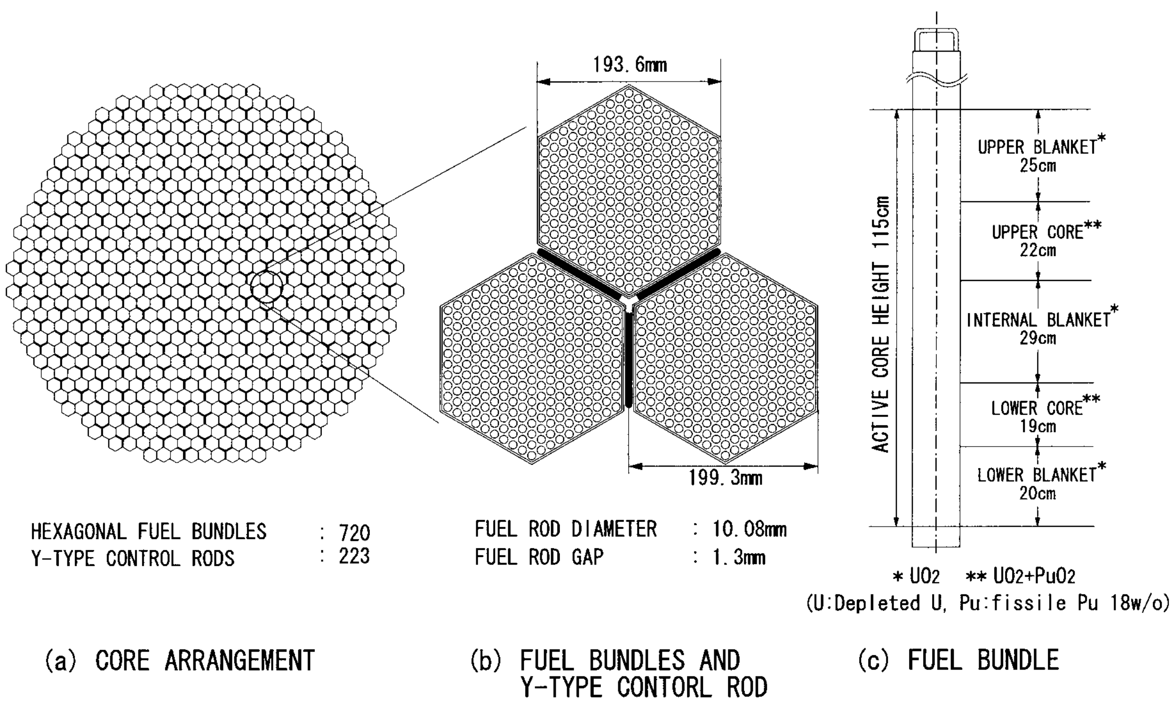

FIG. 4.18-1. RBWR core and fuel design

In an operating BWR, about 85% of the fission reactions occur in the thermal energy region, whereas the medium fission energy is about 7keV in the RBWR-B2, and the reaction rates in the resonance region are comparatively large. Therefore, while a conventional BWR has a Doppler coefficient of -1.6×10^{-5} △k/k/℃, the value in the RBWR-B2 is -3.7×10^{-5} △k/k/℃, or about twice as large. The RBWR-B2 has a void coefficient of -0.5×10^{-4} △k/k/%void compared with -7.0×10^{-4} △k/k/%void for the conventional BWR and so the thermal margin is increased relatively, for example, in the event of pressure elevation or temperature lowering of coolants.

The proposed designs of the RBWR have a negative void coefficient. The typical transients were analysed and the △MCPR, that is, the decrease in thermal margin during transients was evaluated. In the evaluation, the characteristics of the active components in the RBWR were assumed to be those of the ABWR. For example, internal pumps are utilized as recirculation pumps instead of jet pumps in current BWRs. The maximal △MCPR of the RBWR-B2 was 0.14, nearly the same as for current BWRs. The fuel integrity could be maintained because the MCPR was more than 1.3 in the rated power normal operation. The maximum heat flux and the maximum vessel pressure during transients were confirmed as maintained at the limiting values for current BWRs. From a safety perspective, a high pressure flooder break accident, which seems to be the most severe loss of coolant accident (LOCA) for the RBWR, was also analysed. The peak cladding temperature in LOCA was 618℃ for the RBWR-B2, less than the limiting value (1200℃) of current BWRs.

4.18.2.4 Primary components

The Primary system of RBWR is the same as the ABWR except for the core internals and the components of the core cooling system that are effected by the core and fuel design. The RBWR-B2 core was used for this general plant design as this core configuration is is representative of most of the alternative core configurations.

Reactor pressure vessel (RPV)

The diameter of the RPV is about 0.2m larger than the Japanese standard ABWR because the radius of the RBWR-B2 core is about 0.2m larger than the ABWR (2.88 vs 2.69 m) and the downcomer width is 0.1m narrower reflecting rational design margin on RPV irradiation and dimension of RIP (Reactor Internal Pump) installation. The RBWR core is designed to provide the same steam flow and net electrical rating as the ABWR but operates with a higher core void fraction and twice the core exit quality of the ABWR. The height of RPV is about 5m shorter than the ABWR (16 vs 21 m) by considering the shorter core height. Figure 4.18-2 shows the RPV configuration.

Reactor core

The major component changes associated with the use of the RBWR fuel is an about 2.5 meter reduction in the height of the fuel and a similar reduction in the required height of the cavity directly below the reactor where the CRDMs (Control Rod Drive Mechanisms) and CRDM maintenance space is located.

Control rod drive mechanisms (CRDMs)

The number of CRDMs required by the RBWR core increased from 205 for the ABWR to 223. However, the length of the stroke and the lengths of the control blade and associated drive were reduced by about 1.6 meter as shown in Section 4.18.8.

Number and size of internal pumps

As shown in Section 4.18.8, the head requirement is the same for the RBWR core while the total core flow is reduced by 50% due to the higher core outlet quality. However, the 8 RIPs are selected by considering the effect of the transient event of the abnormal 2 RIPs trip.

Steam separators and dryers

The RBWR steam separators must handle the same steam flow, however they operate with one half the total mass flow of the conventional ABWR, therefore further study that is beyond the scope of this initial effort may show that it is possible to reduce the number and/or height of the steam separators. However, no change to the dryer and steam separator geometry's were assumed in this study. Thus, the number, type, and height of the steam separators and steam dryers are assumed to be unchanged form the ABWR.

Supporting safety related equipment

No change in the safety related equipment required to support the RBWR have been made or are contemplated since the size of the shutdown heat removal and core flooding systems are dictated by the thermal rating of the core which is unchanged from the ABWR. Detailed core flooding calculations and re-verification of the core flooding analysis developed for the ABWR with its higher void fraction design will be required to assure that the RBWR fuel will not be unacceptably damaged during Design Basis LOCA.

4.18.2.5 Reactor auxiliary systems

Same as ABWR.

4.18.2.6 Operating characteristics

Same as ABWR.

4.18.3 Description of the turbine generator plant systems

Same as ABWR.

4.18.4 Instrumentation and control systems

Same as ABWR.

4.18.5 Electrical systems

Same as ABWR.

4.18.6 Safety concept

Same as ABWR.

4.18.7 Plant layout

Figure 4.18-3 shows the layout of the primary containment vessel of RBWR, which is the same type containment as ABWR, that is, the RCCV (reinforced concrete containment vessel) and the suppression type configuration. However, due to the high cost of the reactor containment building structures there is a large economic incentive to take maximum advantage of the shorter reactor core and much shorter reactor system to reduce the size and capital cost of the reactor building and the associated containment structure. Since the volumes of the suppression pool (SP) and drywell airspace volume as well as the space required for the steam, feedwater, and core flooding lines and relief valves will limit the height reduction possible with the shorter RBWR core, a conceptual building layout was developed and design basis containment pressure calculated for the revised containment configuration to determine if the reduced suppression pool (SP) and drywell volumes in the shorter containment are adequate given the reduced water inventory and associated stored energy in the RBWR reactor system.

The height of RCCV is about 7m lower than the ABWR (39.9 vs. 33 m) by considering the shorter core height of 2.5m and the reduction of the control rod removal space, the length of the CRDMs, and the height needed for removal and maintenance of the CRDMs. The diameter of RCCV becomes the same as ABWR. The reactor building configuration of RBWR is the same as ABWR but the height is about 7m lower because of the reduction of RCCV height.

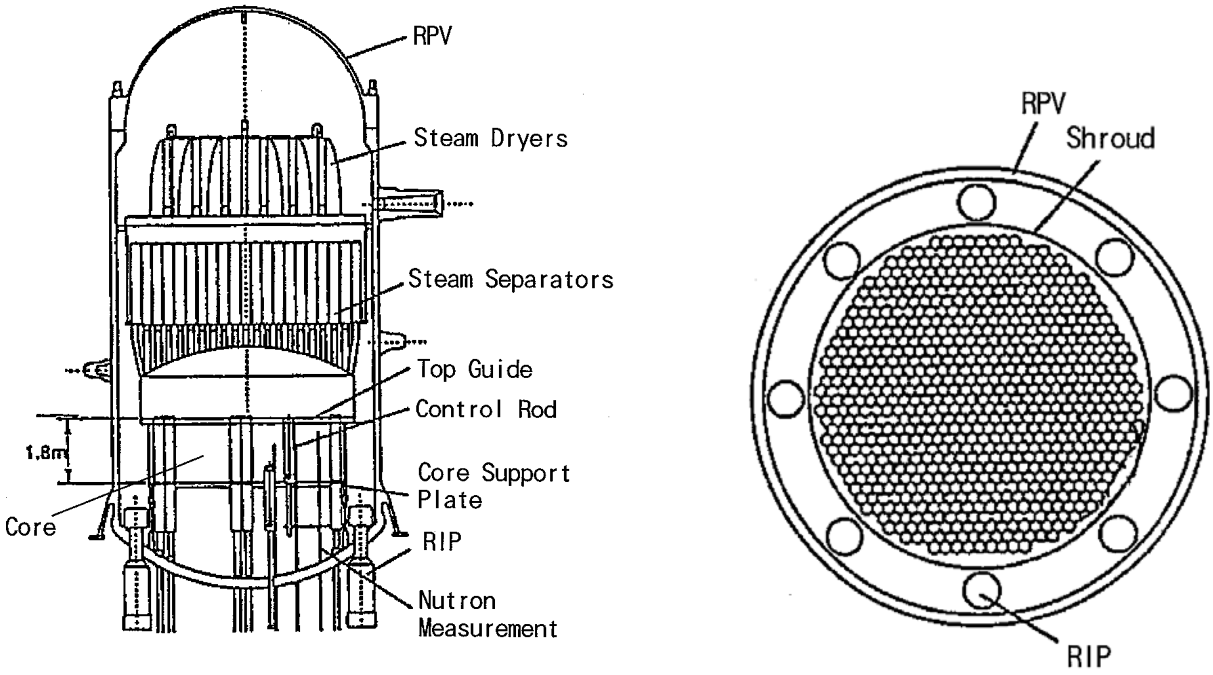

FIG. 4.18-2. Cross-section view of RPV and internal structures of RBWR

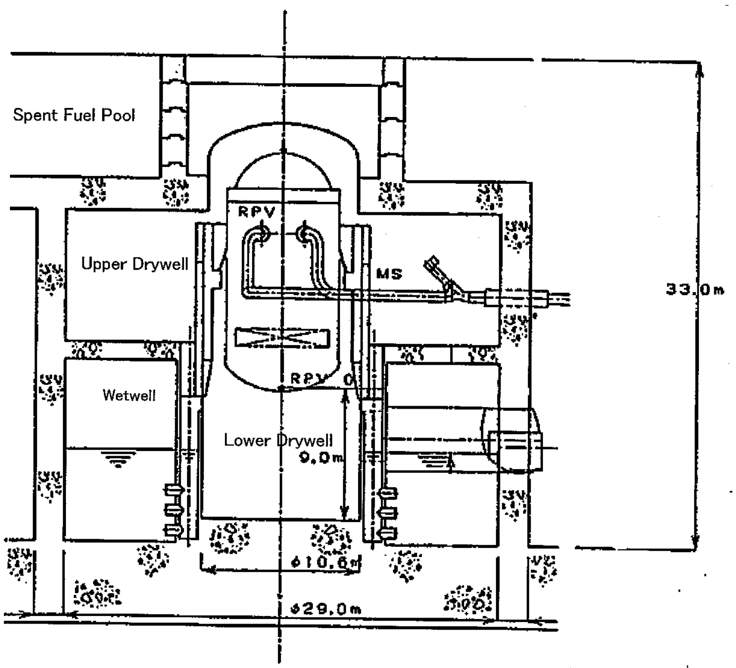

FIG. 4.18-3. Cross-section view of primary containment vessel of RBWR

4.18.8 Technical data (for RBWR-B2)

<u>General plant data</u>		
Power plant output, gross		MWe
Power plant output, net	1356	MWe
Reactor thermal output	3926	MWth
Power plant efficiency, net	34.5	%
Cooling water temperature	≈28.0	℃
<u>Nuclear steam supply system</u>		
Number of coolant loops	1	
Primary circuit volume		m^3
Steam flow rate at nominal conditions	2440	kg/s
Feedwater flow rate at nominal conditions	2440	kg/s
<u>Reacor coolant system</u>		
Primary coolant flow rate	8890	kg/s
Reactor operating pressure	7.17	MPa
Steam temperature/pressure	287.8/7.17	℃/MPa
Feedwater temperature	252	℃
Core coolant inlet temperature	278	℃
Core coolant outlet temperature	288	℃
Mean temperature rise across core	10	℃
<u>Reactor core</u>		
Active core height	1.15	m
Equivalent core diameter	5.57	m
Heat transfer surface in the core	7106	m^2
Heavy metal (Pu+U) weight	126	tWM
Average fuel power density	31.2	kW/kgHM
Average core power density	140	kW/l
Thermal heat flux,Fq		kW/m^2
Enthalpy rise,F_H		
Fuel material	Sintered MO_2	
Fuel(assembly)rod total length	1930	mm
Rod array	triangle lattice	
Number of fuel assemblies	720	
Number of fuel rods/assembly	271	
Number of spacers	5	
Fissile Pu enrichment (range) of first core, average	10.1(appr.)	Wt%
Fissile Pu enrichment reload fuel at equilibrium core	10.7	Wt%
Operating cycle length (fuel cycle length)	14	months
Average discharge burnup of fuel [capability]	>50,000	MWd/t
Cladding tube material	annealed, recystallisedZr2	
Cladding tube wall thickness		
Outer diameter of fuel rods	10.08	mm
Fuel channel/box;material	Zr-4	
Overall weight of assembly, including box	286	kg
Heavy metal (Pu+U) weight/assembly	175	kg
Active length of fuel rods	1150	mm
Number of control rods	223	
Absorber material	B_4C (90% B-10)	
Drive mechanism	electro-mechanical/hydraulic	
Positioning rate	30	mm/s
Soluble neutron absorber		
Breeding ratio	1.01	
<u>Reactor pressure vessel</u>		
Inner diameter of cylindrical shell	7300 mm	
stainless steel		
Wall thickness of cylindrical shell	190	mm
Total height, inside	16000	mm
Base material: cylindrical shell	low-alloy carbon steel	
Design pressure/temperature	8.62/301.7	MPa/℃
<u>Reactor recirculation pump</u>		
Type	variable speed, wet motor, single stage, vertical internal pump	
Number	8	
Design pressure/temperature	same as for RPV	MPa/ °C
Design mass flow rate		

(at operating conditions)	1110 (each)	kg/s
Pump head		
	0.287	MPa
Rated power of pump motor (nominal flow rate)	≪800 kW	
Pump casing material	same as for RPV	
Pump speed (at rated conditions)	<1500	rpm
Pump inertia		kgm2

Primary containment

Type	Pressure-suppression/reinforced concrete	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)		m
Design pressure/temperature	310.3/171.1	kPa/° C
Design leakage rate	0.4 vol	%/day
Is secondary containment provided?	Yes	

Reactor auxiliary systems

Reactor water cleanup, capacity	42.36	kg/s
filter type	deep bed	
Residual heat removal, at high pressure		
at low pressure (100 ° C)	253.8	kg/s
Coolant injection,at high pressure (HPCF)	36.3	kg/s
at low pressure (LPCF)	253.8	kg/s

Power supply systems

Main transformer, rated voltage	24/525	kV
rated capacity	1660	MVA
Plant transformers, rated voltage	24/4.16/13.8	kV
rated capacity	50/15/35	MVA
Medium voltage busbars (6 kV or 10 kV)	13.8/4.16	kV
Number of low voltage busbar systems	3	
Standby diesel generating units: number	3	
rated power	6.57	MW
Number of diesel-backed busbar systems	4160	VAC
Voltage level of these		
Number of DC distributions		
Voltage level of these		
Number of battery-backed busbar systems		
Voltage level of these		

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)		
six flow, tandem compound		
single reheat		
Number of turbine sections per unit (e.g. HP/LP/LP)	1HP/3LP	
Turbine speed	1800	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	6.792/283.7	MPa/℃

Generator

Type	3-phase, turbo-generator	
Rated power	1620	MVA
Active power	1385	MW
Voltage	24	kV
Frequency	50/60	Hz
Total generator mass, including exciter		t
Overall length of generator		m

Condenser

Type	shell type (3 shells)	
Number of tubes	1 tube pass/shell	
Heat transfer area	124,170	m^2
Cooling water flow rate	34.68	m^3/s
Cooling water temperature		℃
Condenser pressure (HP shell)	11.75	kPa

Condensate pumps

Number	4x50%	
Flow rate	435	kg/s
Pump head	3.82	MPa

Temperature		℃
Pump speed		rpm

<u>Condensate clean-up system</u>

Full flow/part flow		full flow
Filter type	(deep bed or rod type)	deep bed

<u>Feedwater tank</u>

Volume		m3
Pressure/temperature		MPa/℃

<u>Feedwater pumps</u>

Number	3x65%	
Flow rate	1000	kg/s
Pump head	6	MPa
Feed pump power		MW
Feedwater temperature (final)	252	℃
Pump speed		rpm

<u>Condensate and feedwater heaters</u>

Number of heating stages,	low pressure	3x4
	high pressure	2x2
	feedwater tank	

4.18.9 Measures to enhance economy and maintainability

The plant economics of the RBWR was evaluated by comparing with ABWR.

The total volume of the RPV is reduced by about 20% because the RPV height is about 5m lower than for the ABWR, though the RPV diameter is about 0.2 m larger for a pancake-like core configuration. The height of the reactor building (R/B) is lower by about 7 m for this RPV compactness, and one floor can be decreased. The number of RIPs is reduced from 10 to 8 because of less required core flow rate. The plant construction cost is, therefore, estimated to be equal or less than ABWR by the effect of decreased amount of material as above-mentioned. Furthermore, the learning effect on the construction cost can be expected because the manufacturing machinery and construction equipment in ABWR are almost available and purchased equipment are compatible with ABWR.

The construction period is not fixed at present since the critical path in its construction was not examined, but it may be shortened due to the one floor reduction in R/B. Several new construction improvements are still going on for ABWR and the challenge to the construction rationalization such as a large block module is easily adopted to RBWR.

The amount of maintenance work and the duration in a periodic maintenance are equivalent to ABWR because the system and equipment are basically the same as ABWR. Learning effect can also be expected as the same maintenance tools are used. Handling of the hexagonal fuel bundles and the Y-type control rods will be the first experience but it is possible to overcome by enough mock-up experiences in advance. In employee dose reduction, a learning effect of ABWR can also be expected.

Finally, the power generation cost will be slightly lower than ABWR from the above-mentioned reasons, excluding MOX fuel cost increase.

4.18.10 Project status and planned schedule

The RBWR design project was initiated in 1980 and the concept of the plutonium generation BWR (PGBR) [Ref .4] was completed, which has a potential for achieving a plutonium generation ratio, i.e., the number of atoms of fissile plutonium left in discharged fuel bundles per that in initial MOX (plutonium-fed natural uranium) fuel bundles, close to 1.0 and better natural uranium utilization by a factor of 5～10 compared with current BWRs. The associated natural uranium utilization and safety margin have been improved by advancing from natural uranium-based MOX fuel bundles and a little bit positive void coefficient of the PGBR in 1988 to depleted uranium-based MOX fuel bundles and negative void coefficient of the RBWR in 1995.

One of the distinctive features of the RBWR is that the effective water-to-fuel volume ratio is smaller than that of current BWRs, so that the calculation method for microscopic epithermal cross sections and the nuclear data library for resonance energy range are important in nuclear characteristic evaluation. For precise neutronic calculations of the fuel bundle designs, the VMONT [Ref.5] code, in which a vectorized Monte Carlo neutron transport method is coupled with the burnup calculation, has been developed. The applicability of the VMONT code to tight lattice configurations was confirmed by the C/E evaluations for k-infinity and reaction rate ratio of 238U capture to 239Pu fission using PROTEUS experimental data [Ref.6]. For thermal hydraulic analysis, the critical power correlation for the tight lattice configuration gains increasingly in importance. So, the modified CISE critical power correlation, which is based on the critical quality-boiling length correlation developed by CISE, has been developed for the application range. Experimental data and calculation results, in which the former data were taken from an experiment on the critical power of the closely packed lattice fuel bundles at Bettis Atomic Power Lab. are used. The proposed correlation reproduces the experimental data with a standard deviation error of 8%.

The RBWR has been designed under proven LWR technology, but it is preferable to confirm a breeding ratio of 1.0 in high burnup fuel bundles, negative void coefficient and mechanical integrity of in-core structure in the same operation condition as that of the 1350 MWe RBWR. A 180 MWth

RBWR has been designed, which has 75 hexagonal fuel bundles, 127 fuel rod per a fuel bundle and the same fuel rod diameter and the fuel rod gap as those of the 1350MWe RBWR, and Hitachi is watching for an opportunity to start constructing the 180 MWth RBWR.

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4.19 CNP1000 (CHINA NATIONAL NUCLEAR CORPORATION, CHINA)

4.19.1 Introduction

The design of the CNP1000 (China Nuclear Power PWR 1000MW) adopts mature and proven technology. The standardized 3 loop 1000 MW class PWR is taken as reference. Some modifications are adopted for improving the performance of the plant. Cost reduction and safety level enhancement are the main objectives of those modifications. Not only shall the current safety requirements be met, but also some modern safety tendencies have been taken into account in the design, for example, some requirements in the URD and the EUR are applied in the conceptual design.

The design goals of the CNP1000 are as follows:

- Power plant output (gross) of 1000 MWe;
- Plant design aimed at a lifetime of 60 years without replacement of the reactor vessel;
- The core is designed with 18-month fuel cycle assuming 87% capacity factor;
- Average fuel batch burnup: 45000 MWd/TU;
- Core design is robust with at least a 15% operating margin on core power parameters;
- Safety target:
  Predicted core damage frequency is below 1E-05/yr
  Frequency of significant release is below 1E-06/yr;
- Occupational radiation exposure is expected to be below 1man –Sv/yr;
- Safety Shutdown Earthquake: 0.2g;
- Unplanned reactor trips are less than one per year;
- Construction period (from pouring the first concrete to commercial operation): 66 months.
-

Compared to the classical 3 loop PWR design, the following modifications are adopted for the sake of safety and economy feature enhancement.

- Number of fuel assemblies and pressure vessel dimensions are increased,
  Number of the CNP1000 FA is 177, compared to classical 157,
  Inner diameter of the CNP1000 pressure vessel is 4340 mm, compared to classical 3989 mm,
  Keep larger thermal margin for the operation flexibilities and realizing 18-month and out-in low leakage fuel cycle,
  Reduction of integral fast neutron flux on vessel for the extension of plant lifetime;
- Containment free volume is increased by increasing the containment dimension,
  To provide larger resistance to pressure shock induced by the large Loss of Coolant Accident (LOCA) and Main Steam Line (MSL) break accident and to keep the containment integrity or delay containment failure after severe accident;
- Engineering safety feature improvement based on preliminary Probabilistic Safety Assessment (PSA) result,
  Higher redundancy for some safety systems, such as EFW pumps (N+2),
  Separation of safety functions and non-safety functions,
  Larger pressurizer water volume and emergency feedwater storage tank volume,
  Layout improvement to assure physical separation of different trains of safety system and amelioration of fire protection;
- Prevention or moderation of severe accidents
  Consideration of countermeasures for Direct Containment Heating (DCH), hydrogen and steam explosion, etc.;
- Digital I&C system
  Digital and computerized I&C system and friendly ergonomic main control room;
- Conventional island
  Half speed turbine-generator set under consideration.

4.19.2 Description of the Nuclear Systems

4.19.2.1 Primary circuit and its main characteristics

The primary circuit of the CNP1000 retains the most general design features of current designs. The reactor coolant system (RCS) consists of the reactor pressure vessel and three loops, each comprising a steam generator, a reactor coolant pump, and reactor coolant pipes. The pressurizer is connected to one of the loops. The reactor coolant system is installed in the containment.

In the reactor coolant system of the CNP1000, compared with a standardized 3 loop PWR, the following considerations are taken into account:

- Change the size of reactor vessel (the diameter of the cylinder was modified from 3989 mm to 4340 mm);
- Enlarge the volume of pressurizer (the volume was enlarged from 39 m^3 to 45 m^3);
- With the function of feed-bleed;
- 60 years lifetime of reactor vessel;
- Reduce the number of welds on the cylinder of the reactor vessel to avoid the weld line located adjacent to the active area;
- Control the chemical composition of reactor vessel strictly;
- Modify the temperature measurement on by-pass line.

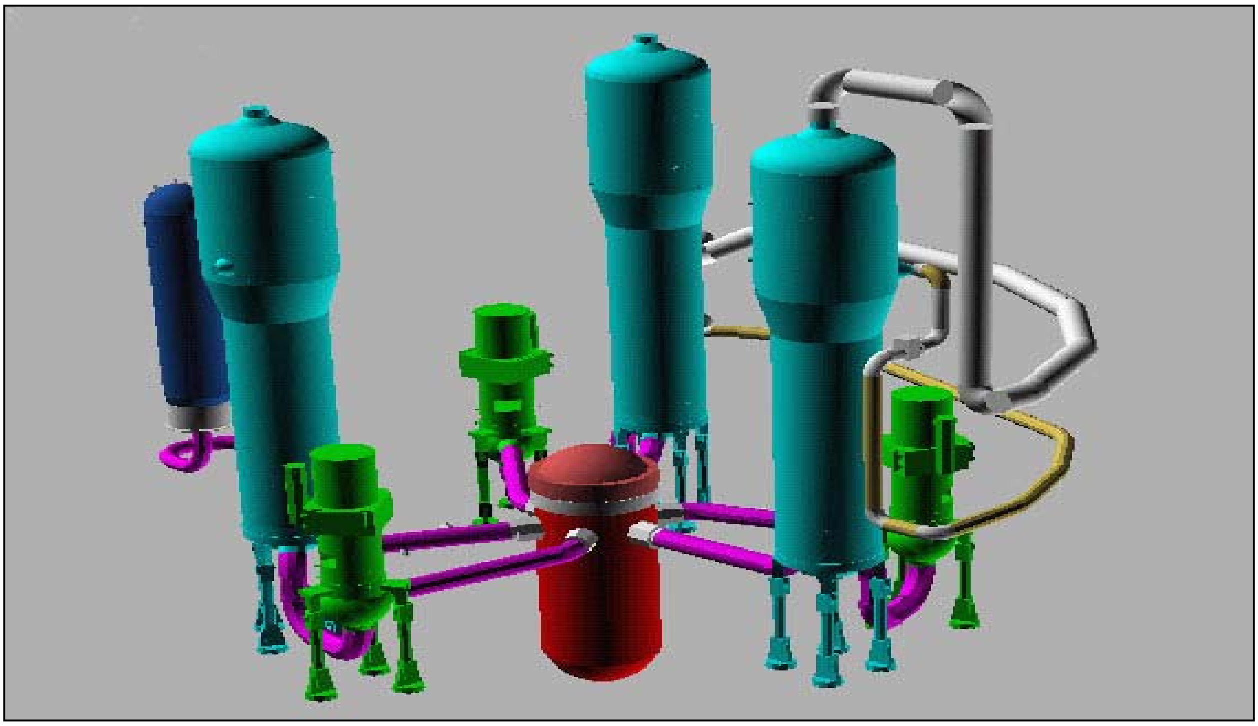


FIG. 4.19-1. CNP1000- Reactor Coolant System

4.19.2.2 *Reactor core and fuel design*

The core of the CNP1000 consists of 177 high performance fuel assemblies.

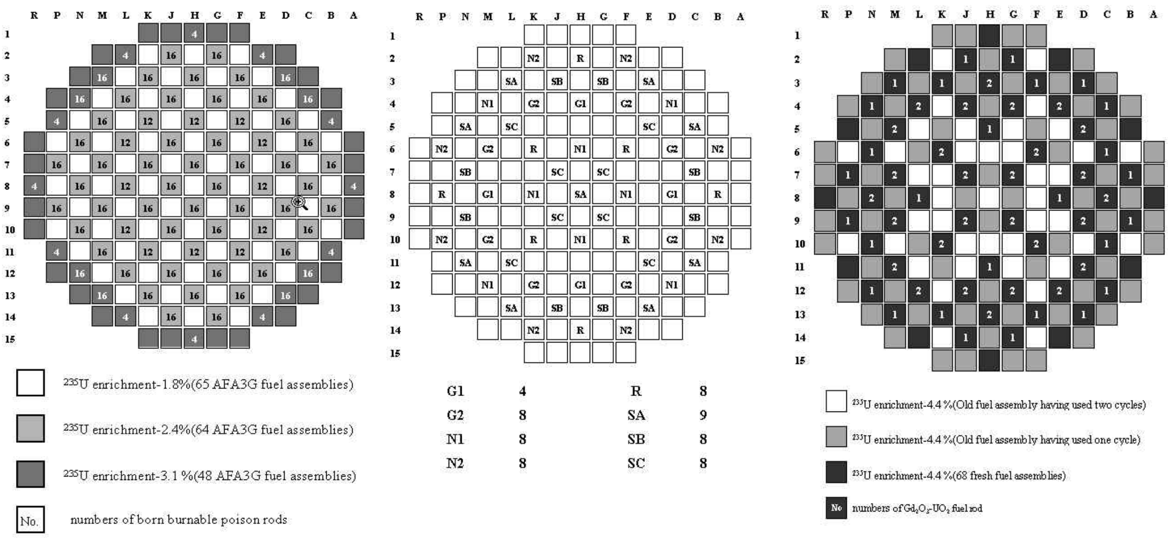
Compared with the standardized 3 loop PWR, the average linear power density of the CNP1000 fuel rods is reduced to 165.0 W/cm, which results in larger thermal safety margins.

Enrichment of the reload fuel at equilibrium core is 4.45%. Based on 1/3 core reload, 475 effective full-power days per cycle can be reached.

The core is designed for a fuel cycle of 18 months with discharge burnup higher.

There are 61 control rod assemblies in the core. According to their functions, they are divided into two groups: control rod group (24 black rod assemblies plus 12 gray rod assemblies) and shutdown rods group (25 black rod assemblies).

The existence of gray rod assemblies increases the operation flexibility, thus the operation mode can be realized



(a) Core loading pattern of the first cycle

(b) Loading pattern of control rod assemblies in core

(c) Loading pattern of the equilibrium cycle

FIG. 4.19-2. CNP1000-Reactor core loading pattern

4.19.2.3 Fuel handling and transfer systems

The fuel handling and transfer system is equipped with a computerized control system to assure the safety of the reloading process, increase the reliability and, consequently, reduce the outage time period.

The CNP1000 units are equipped with 690 high-density spent fuel storage racks, incorporating a composite neutron absorbing material in the rack structure.

The size corresponds to 10 consecutive normal annual discharges plus one complete core unloading. The new fuel storage racks are equipped with 60 bores offering enough space to store one normal annual reload of new fuel.

Fuel enrichment of up to 4.45% is compatible with the design of the spent fuel storage-racks.

4.19.2.4 Primary component

Reactor pressure vessel

The reactor pressure vessel is a cylindrical vessel with a hemispherical bottom head as well as a flanged and gasketed upper head.

The reactor vessel features an integrated reactor vessel head package design, in which equipment mounted on the reactor vessel head has been rearranged so that it can be removed as a single assembly. As a result, the reactor vessel head can be opened and closed easily with shortened disassembly time and reduced radiation exposure.

Consistent with the core size of 177 fuel assemblies, the inner diameter of the reactor vessel is 4340 mm.

To realize a design target of 60 years lifetime, following measures are taken:

- Base material with low Cu and P, resulting in significant reduction in the reference nil ductibility transition temperature (RTNDT) at the end of life, thus increasing the margins with respect to brittle fracture;
- Reducing the core linear power density, decreasing the neutron flux on the inner wall of the vessel;
- Large water clearance and low leakage program also reduces the neutron flux on the inner wall of the vessel.

Reactor internals

The reactor internals guide the reactor coolant flow from the inlet nozzles downward, then up through the reactor core.

Steam generator

The steam generator is a vertically mounted U-shaped tabulate heat exchanger.

The primary channel head is welded to the tube sheet. The surface in contact with the primary coolant is clad with Inconel or stainless steel.

The U-tubes are expanded into the tube sheet and seal welded to the tube sheet cladding.

All pressure boundaries are made of low-alloy steel except the Inconel tubes.

Inconel 690 TT (Thermally Treated) is the tube material to increase resistance to intergranular and stress corrosion cracking.

The thermal transfer area of steam generator is 5630 m^2.

Pressurizer

The pressurizer is used to control and adjust the pressure of the reactor coolant system. On the upper head of the pressurizer, pilot safety valves are equipped to provide overpressure protection.

The total volume of the pressurizer is 45 m^3. This large pressurizer of the CNP1000 increases the capability of the transient response.

Reactor coolant pump

The reactor coolant pump is a vertical, single-stage, shaft-seal, mixed flow type pump.

The pump shaft is coupled to the motor. The primary coolant is drawn up through the casing bottom and pressurized by the impeller and diffuser, and is delivered through the horizontal discharge nozzle. The thermal barrier assembly is placed behind the impeller to eliminate thermal conduction from the primary coolant to the bearing and the shaft seal section.

The model 100 pump incorporates improvements such as: radial-discharge, single piece casing, and diffuser incorporated into the hydraulic stage. The result is that the pumps are more compact and more efficient. For example,

- The hydraulic design for the impeller-diffuser casing provides a higher hydraulic efficiency;
- The thermal barrier enclosure is bolted instead of welded, and is removable;
- The cartridge design of the shaft seal system has also been improved, to provide easier access during inspection and maintenance and to reduce the replacement time.

4.19.2.5 Reactor auxiliary systems

Chemical and volume control system (CVCS)

The chemical and volume control system is a non-safety related system. The charging pumps no longer undertake a high pressure injection function.

Residual heat removal systems (RHR)

The residual heat removal system is used to remove heat energy from the core and the reactor coolant system during the plant shutdown after an initial cooldown and depressurization through the steam generators.

The RHR system is used for:

- Reducing the reactor coolant temperature to the shutdown value;
- Maintaining the reactor coolant temperature at cold shutdown value;
- Providing forced flow of coolant through the core;
- Transfer of refueling water;
- Serving as a line for reactor coolant purification.

The main equipment includes 2 heat exchangers, 2 pumps, 2 series of pipe lines equipped with flow control valves and safety valves. It is divided into 2 trains, with each train consisting of one 100% capacity motor-driven pump and one 100% heat exchanger.

RHR pumps are located in the reactor building in order to prevent radioactivity from intersystem LOCA.

4.19.2.6 Operating characteristics

The CNP1000 uses an operating mode with the following performance:

- Operates under stable continuous base load conditions;
- Performs load follow according to a 12-3-6-3 daily load cycle, by which it can operate at rated power for 12 hours, at 30% of rated power for 6 hours, the transition periods lasting 3 hours;
- Perform step load changes of +10% of rated power and ramps load changes at +5% of rated power per minute;
- Performs load rejection from full load to house load without reactor trip;
- Operates under remote frequency control (also known as load regulation).

This kind of proven operating mode provides:

- A large operating margin allowing a low likehood of trips and a high operability of the plant;
- A significant safety margin available for unexpected operating occurrence.

4.19.3 Description of turbine generator plant systems

The main systems in the conventional island of the CNP1000 includes: main steam system, main feed water system, condensate extraction system, moisture separator-reheater system, turbine by-pass system, start-up feedwater system and auxiliary steam system, etc.

The main components of the turbine generator plant system include: turbine, generator, condenser, moisture separator-reheater, high pressure heater, low-pressure heater, deoxidizer and water tank, condenser pump, main feedwater pump and related instrumentations, valves and pipes.

4.19.3.1 Turbine generator plant

The turbine generator is designed as a half-speed set. A digital electric-hydraulic adjusting system is utilized for the turbine generator set.

4.19.3.2 Condensate and feedwater systems

They are the same as that of standardized 3 loop PWR.

4.19.3.2 Auxiliary systems

They are the same as that of the standardized 3 loop PWR.

4.19.4 Instrumentation and control systems

The I&C system provides various kinds of control, protection and monitoring information and measures for the nuclear system and components, thus to ensure the safety, reliability and economy of the nuclear power plant.

The Integrated Digital I&C System of the CNP1000 comprises all the electrical, electronic and pneumatic equipment for the function of information, control and protection as well as other auxiliary functions.

4.19.4.1 Design concept, including control room

The design principles of the I&C system are as follows:

- Function requirements:
  Basic requirements refer to existing 1000MW PWR
  Considering the enhance capability.

The function of the digital I&C system can be extended to make the operation more flexible and safer.

- The system is split into two classes: safety class system and non-safety class system;
- For system architecture, it is split into four levels corresponding to the main function
  process interface（level 0）
  automation（monitoring, control and protection） (level 1）
  supervision and control （unit level）（level 2）
  site management and remote management and configuration（plant）(level 3);
- For the main control room, partially using state-oriented operation procedure;
- For operator organization: there are one supervisor and two operators for each unit, and one safety engineer for two units;
- Using proven and modern computer technology, redundant net structure and modularized hardware structure.

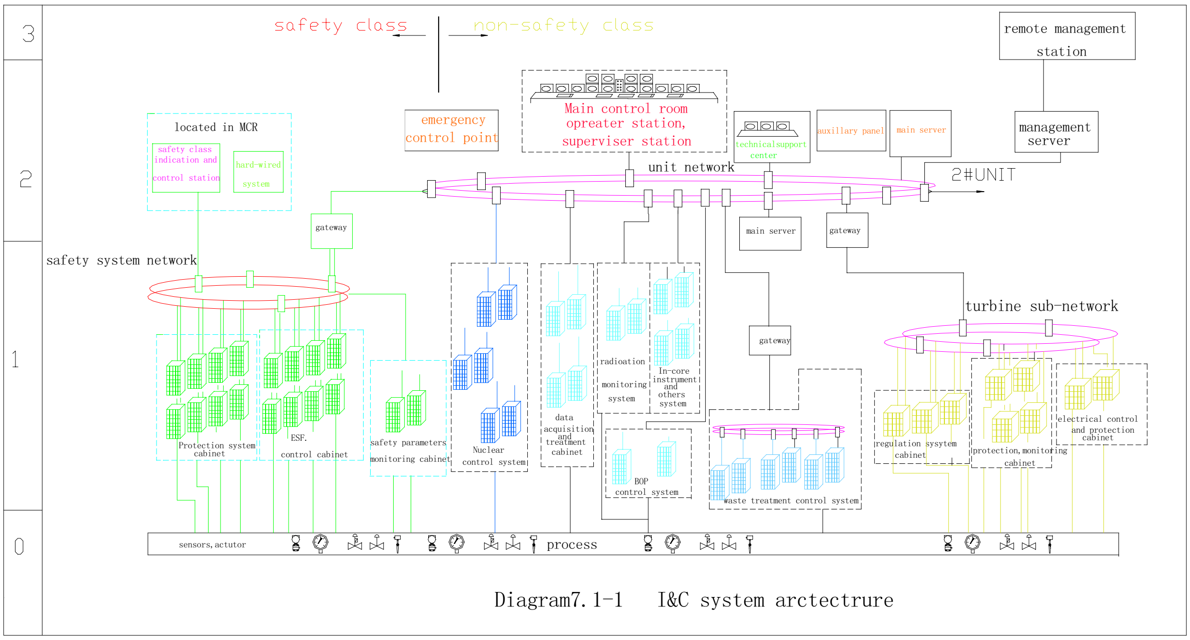


Diagram7.1-1 I&C system arctectrure

FIG. 4.19-3. CNP1000- I&C system architecture

Main control room

The functions of the main control room are as follows:

- To provide man-machine interface;
- To support the operator in getting knowledge of the plant condition;
- To help the operator in making a strategy;
- To implement control action.

The design principles of the main control room are as follows:

- Objective of the control room;
- Organization of the operating staff;
- Human factor engineering;
- Safety principle.
- Functional analysis,
- Radiation protection,
- Fire proofing,
- Aseismatic.

The I&C system also includes:

- Computerized workstation
  Three identical workstations
  The operator can rapidly access all necessary information from his workstation;

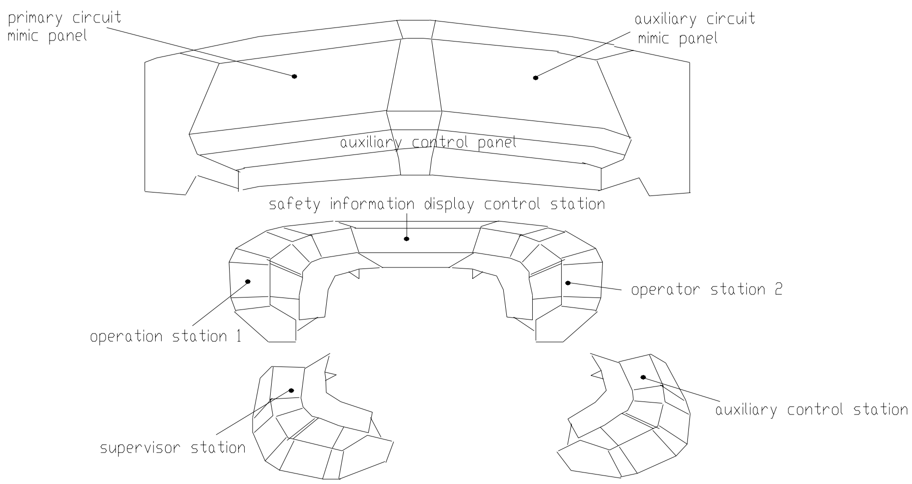


FIG. 4.19-4. CNP1000- Main Control Room

- Safety indication and control station
  Safety display and control equipment
  Hardwired pushbutton;
- Mimic panel
  To give the operator an overall view of the plant
  To provide a common reference of the plant status to all members
  To permit personnel who are not permanently stationed in the control room to get a rapid global view of the plant process
  To provide redundant means to display information;
- Auxiliary panel
  Commands, displays and indicators are directly hardwired to level 1.
  To get the installation to safe shutdown in case of failure of the workstations;
- Emergency shutdown panel
  Two workstations which belong to train A and train B. Their functions are the same as of those workstations in the control room.
  Two panels which belong to train A and train B are hardwired to the level 1.
  The emergency shutdown panels are locked when the main control room is available;
- Technical support center
  One workstation, where specific staff can monitor the process of the plant, make recommendations to the main control room shift members.

4.19.4.2 Reactor protection and other safety systems

Reactor protection system

The reactor protection system automatically generates emergency shutdown and safety system driven to limit the consequence of the accident when the reactor operation exceeds the safety limitation, to ensure the safety of the reactor and components, people and environment. Furthermore, the reactor protection system can also inform operation staff of the plant operating state in time and provides the safety-related signal to the safety monitoring system.

The reactor protection system is composed of in-core and ex-core measure systems, a corresponding process monitoring system, reactor trip system, reactor trip mechanism and ESF.

The system characteristics are as follows:

- Microprocessor, distributed system construction;
- Automatic and permanent hardware self testing;
- On-line DNBR and LPD calculation;
- Open communication capabilities.

The system architecture is as follows:

- Protection parameter process subsystem,
- Protection logic process subsystem,
- Protection actuating subsystem,
- Protection testing subsystem,
- Information processing subsystem,
- Network.

The improved precept of the protection system is as follows:

- Using digital technology, all processing and calculations are based on software,
- Using full two-out-of-four logic.

4.19.5 Electrical systems

To enhance the capability against Station Blackout (SBO), the following countermeasures are taken in the CNP1000 design:

- Equip the residual steam driven system to ensure the operation of hydraulic test pump, the power supply of essential control and instrumentation as well as lighting;
- Provide emergency injection for the seal of main coolant pumps to avoid Seal LOCA;
- Enlarge the storage of compressed gas to prolong the operating time for all air-operated valves;
- Establish an emergency procedure to cope with SBO. The emergency procedure is provided as guidance for operators to control units into safety state after SBO occurs;
- Add Alternate AC Source (AAC) power supply;
- Enlarge the capacity of storage battery.

4.19.5.1 Operational power supply systems

They are the same as that of the standardized 3 loop PWR.

4.19.5.2 Safety-related systems

They are the same as that of the standardized 3 loop PWR.

4.19.6 Safety concept

4.19.6.1 Safety requirements and design philosophy

The engineered safety systems are designed to meet the single-failure criterion. The general organization and design of the safety systems submit to the “two-trains” concept, (each system is composed of two totally separate redundant sub-systems or “trains”), plus diversified backups to cope with beyond design-basis accidents.

To ensure the safety function and also to simplify the design, normal operating function by the safety systems are no longer undertaken. More detailed information is described in the following sections.

The engineering safeguard safety systems include:

- Emergency Core Cooling System (ECCS),
- Containment Spray System (CSS),
- Emergency Feedwater System (EFWS).

4.19.6.2 Safety systems and features (active, passive and inherent)

Emergency Core Cooling System (ECCS)

The functions of the emergency core cooling system are as follow:

- To make up water to the reactor coolant system (RCS) following a LOCA to assure that sufficient water inventory is maintained to permit adequate core cooling;
- To inject boron water into the RCS to achieve cold shutdown.

The ECCS system consists of:

- 2x100% high head safety injection (HHSI) pumps;
- 2x100% low head safety injection (LHSI) pumps;

- 3 accumulators filled with 2100 ppm borated water;
- 1 borated water tank which is filled with 7000 ppm borated water;
- 1 set of hydraulic test pump.

The adoption of low concentration borated water (7000 ppm) avoids the jam in the injection pipes due to the crystalization of boron.

By this modification, the suction head of HHSI pumps is reduced, and the HHSI pumps can operate independently without LHSI pumps. This modification leads to higher system simplification, higher operation flexibility and efficiency and, finally, higher reliability.

Containment Spray System (CSS)

The functions of the containment spray system are as follows:

- To reduce containment pressure and temperature to acceptable levels for maintaining containment integrity under accidental conditions that cause an increase in pressure and temperature of the containment (LOCA or steam line break inside the containment);
- During a LOCA , the system also decreases the airborne radioactivity level (especially iodine) within the containment;
- Can be used, about 15 days after LOCA in case of the failure of the ECCS low head safety injection pumps by special procedure operation.

Each plant unit consists of two identical trains A and B, which are physically separated.

Each train consists of:

- One pump, one heat exchanger, two spray headers located at two different levels in the dome, the pipes connecting the various equipment, a pump test line;
- One sump, a pump-sump connection and a pump-refueling water storage tank (RWST) connection.

Compared to the standardized 3 loop PWR design, the chemical additive tank which is filled with a NaOH solution is eliminated in the CNP1000 design. This modification is aimed to reduce contamination of the equipments and pipes which are located in the containment after un-normal spray occurrence. Another kind of chemical substance such as trisodium phosphate dodecahydrate is used in place of the NaOH solution for containment spray water pH control.

Emergency Feedwater System (EFWS)

The EFW system is classified as an engineered safety system. The functions of the emergency feedwater system are as follows:

- In the event of failure of one of the normal feedwater systems, the EFW system operates to remove residual heat until the conditions for placing the residual heat removal system in operation are reached;
- The heat removed from the reactor coolant system is transferred to the secondary system via the steam generators supplied by the auxiliary feedwater system; the secondary system itself is cooled by the condenser or atmospheric steam dump system.

The emergency feedwater system is composed of two redundant trains. Each train includes 1 motor driven pump and 1 turbine driven pump, some relevant pipes and valves, etc.

Compared with the standardized 3 loop PWR design, there are some improvements on the EFWS:

- Adds an independent startup feedwater system, separates the normal operating function and safety function;
- The Startup Feedwater System serves for supplying feedwater to the secondary side of the steam generators, during startup of the Reactor Coolant System, hot shutdown and cold shutdown;
- Four pumps system: there are 4 pumps (2 motor pumps and 2 turbine pumps). Safety improvements are as follows:
  Redundancy of pumps is N＋2,
  Reliability of the system is increased, therefore, the probability of core melt is reduced.
- Adds another storage tank and changes the volume of storage tanks to 600 m^3. There are connection pipes and valves between the two tanks. Normally each tank is independent.

4.19.6.3 Severe accidents (Beyond design basis accidents)

In order to reach a higher safety target (Predicted core damage frequency is below 1E-05/yr, and frequency of significant release is below 1E-06/yr) than that of the existing nuclear power plant, severe accident management is considered as an important portion in the design of the CNP1000.

To prevent design basis accidents from leading to severe accidents, the CNP1000 takes the following measures:

- Increased core thermal margin;
- Improved reliability of safety systems and equipments;
- Simplified design of the important safety systems while retaining redundancy and diversity;
- Optimized man and machine interface and digital I&C systems;
- Constituted special procedures which cope with the important beyond design basis accident sequences;
- Constituted state- oriented accident procedures;
- Countermeasures such as feed and bleed function, prevention of high-pressure meltdown and DCH are under consideration.

To strengthen severe accident monitoring, compared with standardized 3 loop PWR, the CNP1000 has following improvements:

- Instrumentations are installed to monitor core uncovery, melt down of the reactor vessel lower head and the fuel debris which drops down to the bottom of the cavity. Monitoring measures include water level measurement in the pressure vessel and the temperature measurement for outside wall of the pressure vessel lower head as well as lower wall of the reactor cavity;
- Besides the narrow measure range inside the containment, a wide range pressure instrumentation used for the condition of severe accident and radiation monitoring instrument will also be installed;
- For prevention or delay of the containment failure, the following measures are taken:
  Larger containment free volume with higher pressure shock resistance.
  Countermeasures for hydrogen and steam explosion are under consideration.

The severe accident management guidance is established.

4.19.7 Plant layout

4.19.7.1 Buildings and structures, including plot plan

The general arrangement of the CNP1000 is based on a twin-unit concept. This arrangement enables two units to share certain auxiliary systems that are not important for the safety of individual plant units. It ensures the compactness of the building which results in saving in investment costs and reducing operating costs.

The CNP1000 consists of the nuclear island and turbine island. The nuclear island includes the reactor building, nuclear auxiliary building, emergency diesel generator building, fuel building and electrical building. A plot indicating the relative location of structures within CNP1000 scope is shown in figure 4.19-5, and 4.19-6. The layout complies with the physical separation of equipment for the safety systems. The arrangement is designed to ensure that an incident or fortuitous initiating event does not propagate or contribute to an accident, the consequences of which could be greater than for the original event.

The layout satisfies the safety requirements of radiation protection, internal missiles, fire, internal flooding. The arrangement is also designed for the convenience of maintenance, considering accessibility to and replace ability of equipment.

The design strength of nuclear safety-related building is based on Safe Shutdown Earthquake value.

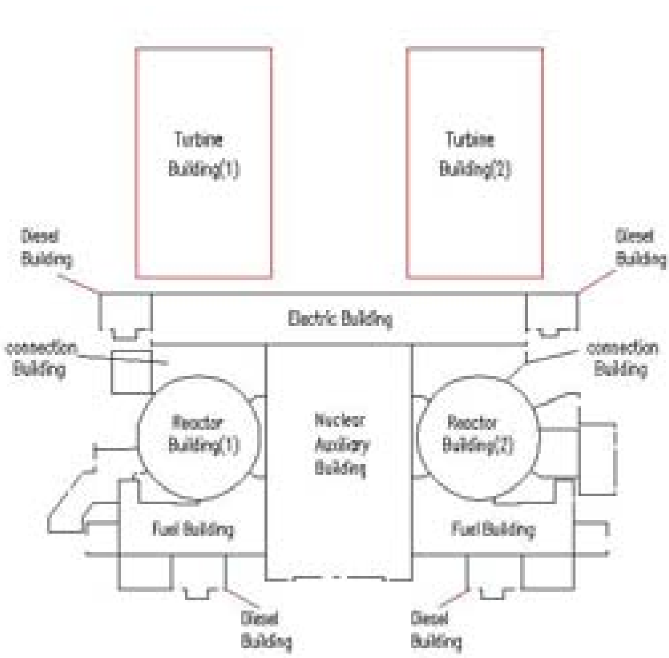


FIG. 4.19-5. CNP1000 - Plant general arrangement

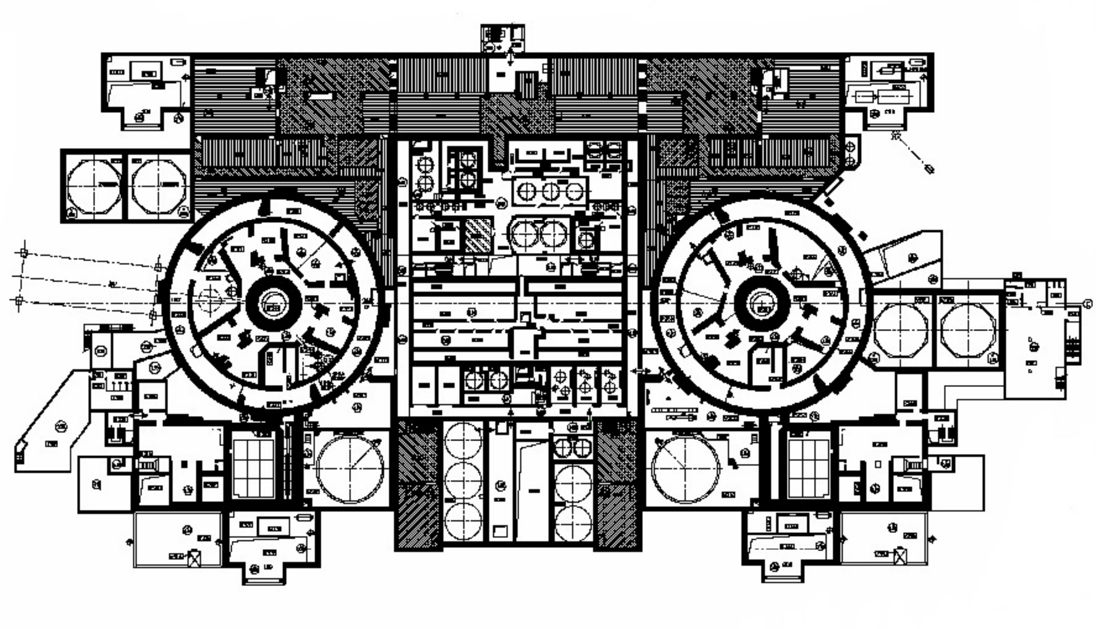


FIG. 4.19.-6 CNP1000 – Plant general arrangement plan at EL. + 5.00 m

4.19.7.2 Reactor building

The reactor building is the central building of the plant. It essentially coincides with the containment building. The reactor building is designed to ensure physical separation against the external and internal missiles. It is taking into account seismic loads and the effects of severe accidents.

4.19.7.3 Containment

The containment building is designed to contain the release of airborne radioactivity following postulated design basis accidents and to provide shielding of the reactor core and reactor coolant system during normal operations. A seismic category I structure, the containment building is a free standing single, cylindrical shape in prestressed concrete with an oblate roof, and steel liner. The containment building houses the reactor core, reactor coolant system, reactor refueling cavity, and portions of the safety feature and auxiliary systems.

The containment building is designed to provide biological shielding, external missile protection, and to sustain all internal and external loading conditions which may reasonably be expected to occur during the life of the plant. The interior arrangement of the containment building is designed to meet the requirement for all anticipated conditions and maintenance.

4.19.7.4 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 classed as non-safety related.

4.19.7.5 Other buildings

Fuel building

The fuel building is designed to provide for the handling, transferring and storage of new or spent fuel. The fuel building is protected against external hazards, such as earthquake, aircraft crash and explosion pressure wave.

Auxiliary building arrangement

The auxiliary building is designed to share by two plant units, and houses mainly:

- Chemical and volume control system,
- Reactor boron and water make-up system,
- Boron recycle system,
- Steam generator blowdown system,
- Common radioactive waste storage and treatment systems.

The auxiliary building is divided into six areas, and seven floor levels. It is a seismic category I structure.

Emergency diesel generator building

There are four emergency diesel generator buildings located in the four corners of the nuclear island building. These building are seismic category I structures which provide protection from fire, missiles, and the environment. Each building contains an emergency diesel generator and its auxiliary equipments.

4.19.8 Technical data

General plant data

Power plant output, gross	1000	MWe
Power plant output, net		MWe
Reactor thermal output	2895	MWt
Power plant efficiency, net		%
Cooling water temperature		℃

Nuclear steam supply system

Number of coolant loops	3	
Primary circuit volume, including pressurizer		m^3
Steam flow rate at nominal conditions		kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	/6.76	℃/MPa
Feedwater temperature/pressure	230/	℃/Mpa

Reactor coolant system

Primary coolant flow rate[20.833 m^3/s]	14 668	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	292.1	℃
Coolant outlet temperature, at RPV outlet	325.9	℃
Mean temperature rise across core		℃

Reactor core

Active core height	3.658	m
Equivalent core diameter	3.228	m
Heat transfer surface in the core		m^2
Total fuel inventory [UO_2]	81.6	t
Average linear heat rate	16.50	kW/m
Average fuel power density	35.4	kW/kg U
Average core power density (volumetric)	95.1	kW/l
Thermal heat flux, F_q	597	kW/m^2
Enthalpy rise, F_H	1.65	
Fuel material	UO_2, UO_2+Gd_2Q_3	
Fuel assembly total length		mm
Rod array	17×17 AFA-3G	
Number of fuel assemblies	177	
Number of fuel rods/assembly	264	
Number of guide tubes for control rods/instr.	24/1	
Number of spacers		
Enrichment (range) of first core	1.8, 2.4, 3.1	Wt%
Enrichment of reload fuel at equilibrium core	4.5	Wt%
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel		
Cladding tube material	Zr-4	
Cladding tube wall thickness	0.62	mm
Outer diameter of fuel rods		
Overall weight of assembly		kg
Active length of fuel rods		
Burnable absorber, strategy/material	Gd_2Q_3, mixed with fuel	
Number of control rod assemblies	61 (49 black + 12gray rods)	
Absorber rods per control assembly		
Absorber material, black/gray rods	Ag-In-Cd / stainless steel	
Drive mechanism	Magnetic jack	
Positioning rate		mm/min（or mm/s）
Soluble neutron absorber		

Reactor pressure vessel

Cylindrical shell inner diameter	4340	mm
Wall thickness of cylindrical shell	220	mm
Total height	12 547	mm
Base material: cylindrical shell	A 308-3	
RPV head		
Design pressure/temperature	17.2/343	MPa/℃
Transport weight(lower part) [include. head]		t
RPV head		t

Steam generator

Type	Vertical, U-tube heat exchanger	
Number	3	
Thermal transfer area	5630	m^2
Number of heat exchanger tubes	6307	
Tube dimensions(O.D/thickness)	19.05/1.09	mm
Maximum outer diameter		mm
Total height	21 000	mm
Transport weight	365	t
Shell and tube sheet material		
Tube material	Inconel 690	

Reactor coolant pump

Type	Single-stage centrifugal pump with canned motor	
Number	3	
Design pressure/temperature	17.2/343	
MPa/℃		
Design flow rate (at operating conditions)[6.944 m^3/s]	5 146	kg/s
Pump head	100	mH_2O
Power demand at coupling, cold/hot	9 060/6 800	kW
Pump casing material	Stainless steel	
Pump speed	1500	rpm

Pressurizer

Total volume	45	m^3
Steam volume: full power/zero power	18/	m^3
Design pressure/temperature	17.2/360	
MPa/℃		
Heating power of the heater rods	1440	kW
Number of heater rods	63	
Inner diameter	2166	mm
Total height	14500	mm
Material	A508-3	
Transport weight		t

Pressurizer relief tank

Total volume		m^3
Design pressure/temperature		MPa/℃
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	single, prestressed concrete with steel liner	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	39/66.48	
Free volume	56 040	m^3
Design pressure/temperature(DBEs)	550/-	kPa/℃
(severe accident situations)	550/-	kPa/℃
Design leakage rate		vol%/day
Is secondary containment provided?		

Reactor auxiliary systems

Reactor water cleanup,	capacity	kg/s
	filter type	
Residual heat removal,	at high pressure	kg/s
	at low pressure	kg/s
Coolant injection,	at high pressure	kg/s
	at low pressure	kg/s

Power supply systems

Main transformer,	rated voltage	kV
	rated capacity	MVA
Plant transformer	rated voltage	kV
	rated capacity	MVA
Start-up transformer	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		MW
Number of diesel-backed busbar system		
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 dc

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)		
Number of turbine section per unit(e.g. HP/LP/LP)		
Turbine speed	1500	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/℃

Generator

Type	
Rated power	MVA
Active power	MW
Voltage	kV
Frequency	Hz
Total generator mass	t
Overall length of generator	m

Condenser

Type	
Number of tubes	
Heat transfer area	m^2
Cooling water flow rate	m^3/s
Cooling water temperature	℃
Condenser pressure	kPa

Condensate pumps

Number	3×50%	
Flow rate		kg/s
Pump head		
Temperature		℃
Pump speed		rpm

Condensate clean-up system

Full flow/part flow

Filter type

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/℃

Feedwater pumps

Number	
Flow rate	kg/s
Pump head	
Feedwater temperature	℃
Pump speed	rpm

Condensate and feedwater heaters

Number of heating stages

Redundancies

4.19.9 Measures to enhance economy and maintainability

No information provided.

4.19.10 Project status and planned schedule

No information provided.

CHAPTER 5. MEDIUM SIZE ADVANCED LWR DESIGNS (300–700 MW(e))

5.1 AC-600 (CHINA NATIONAL NUCLEAR CORPORATION, CHINA)

5.1.1 Introduction

The design of the AC-600 (Advanced Chinese PWR) is based on the design of the Qinshan phase II PWR nuclear power plant (2×600 MWe) with the objective of improving the economy and safety of the nuclear power plant （NPP） through use of system simplification, passive safety, and modular construction. The AC-600 is expected to become a major type of reactor for the next generation of 600 MWe nuclear power plants in China.

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 so as to function as accident mitigation. The AC-600 design, eliminating 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 systems very much, largely improving the safety of AC-600. The major design targets of AC-600 are given in Table 5.1-I.

The safety goals of nuclear power plants should include not only the protection of the environment and the public, but also 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 the goals of protecting the environment and public health can only be attained under the prerequisite of the safety of the nuclear power plants themselves. Increasing the plant's own safety and preventing core melt should be emphasized so as to restore the public confidence in nuclear power.

TABLE 5.1-I. MAJOR DESIGN TARGETS FOR THE AC-600

Parameter	Design Target
Core melt frequency	$<1\times10^{-5}$ to 1.5×10^{-6}/r-y
Maximum individual dose on site boundary after severe accidents whose frequency is less than 1×10^{-6} /r-y	<0.25Sv
Operator response time for DBA	72 hours
Availability factor	≥ 90%
Plant life time	60 years
Emergence trip frequency	Less than once a year
Plant personnel exposure dose	<0,5-1.0 man-Sv/year
Refuelling period	18-24 months
Construction cost	about 20% less than current operating 600 MWe plant
Construction period	48 to 60 months

5.1.2 Description of the nuclear systems

5.1.2.1 Primary circuit and its main characteristics

The AC-600 reactor plant design is based on the design of Qinshan phase II, but it incorporates a number of improvements and safety enhancements compared with the plants of Qinshan phase II. The primary circuit of the AC-600 uses 2 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 (Figures 5.1-1 and 5.1-2).

In the AC-600 design, the vertical distance between the steam generators and the reactor core has been increased, and the flow resistance of the coolant loops have been reduced, 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 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 leads to a large safety margin for loss of flow accidents and has a benefit to the characteristics of the secondary Passive Residual Heat Removal System.

5.1.2.2 Reactor core and fuel design

The reactor core consists of 145 17×17 advanced fuel assemblies (AFA-3G), 57 control rod assemblies and other associated assemblies. There are 45 black rod (Ag-In-Gd) and 12 grey rod (stainless steel) control assemblies in the core.

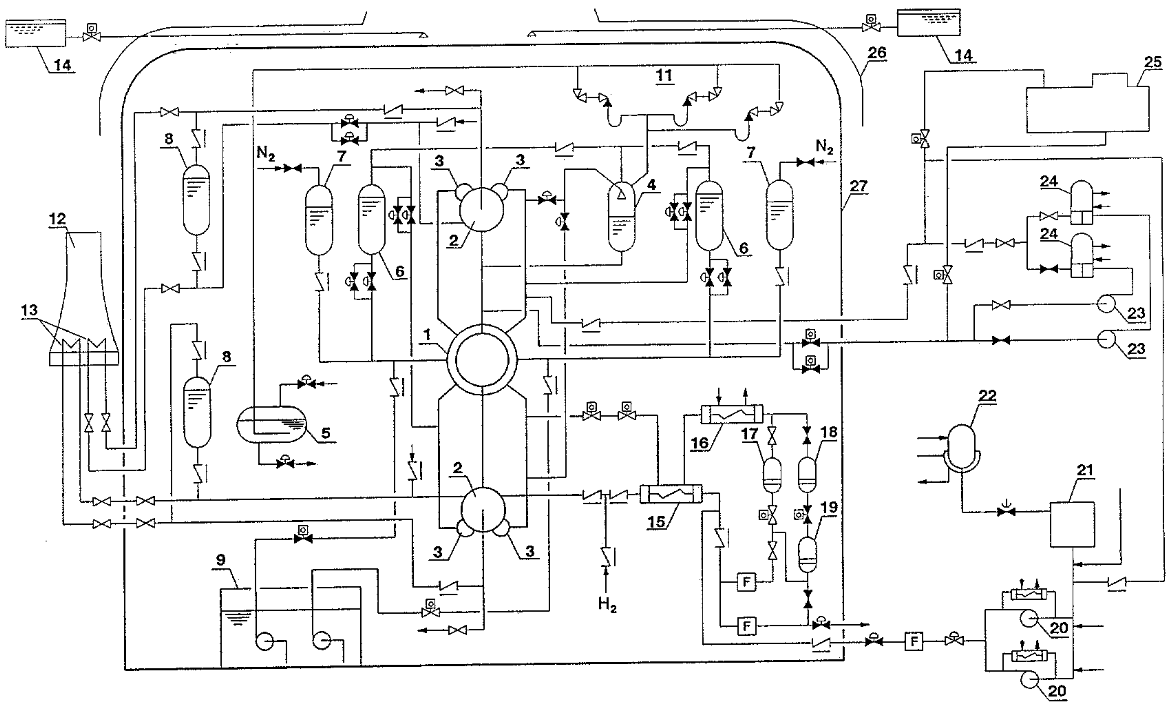
The technology for automatically searching optimum low leakage in-out refueling strategy has been employed in reactor core and fuel management design. This technology contributes to increase average batch discharge burnup (more than 45000 MWd/tU), to extend the refueling period and to reduce the fuel cycle cost.

The reactor core of AC-600 is a low power density design whose average linear power density is 13.42 kW/m(meet the design requirment of URD), and the minimum DNBR is 2.45 (design limit is 2.0). The small core power density results in large thermal safety margins for normal operation and accident conditions (>15%).

The AC-600 design uses Gd_2O_3 burnable poison, solidly melted in the fuel, to reduce the excess reactivity of the reactor and the critical boron concentration. Due to 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.

A kind of radial iron-water neutron reflector is adopted to take the place of abaffle and heat shielding assembly. The test in reactor under the conditions of pure water and borated water and the caculational results of computer code demonstrate that:

- The attenuation rate of fast neutron flux, accquired in a reactor core with reflector, is larger than that accquired in a no-reflector core. This result shows that the reflector design adopted in AC-600 can effectively reduce the damaging fluence of fast neutrons on the RPV and extend its service life .
- Core reactivity can benefit from this kind of reflector, and when the volume of iron and borated water is in the proportion of 9:1, the reflector is of the most significant benefit .
- The hydraulic analysis of AC-600 is performed by 3D-flow field computer code, and astatistical method is also applied to improve the thermal-hydraulic design. Thanks to the improved method, the calculation result is more reasonable and more compatible with the result of reactor flow field distribution optimization tests.



Legend:

Item	Name	Quantity	Item	Name	Quantity	Item	Name	Quantity
1	reactor	1	10	low pressure safety injection pump	4	19	cation bed exchanger	1
2	steam generator	2	11	safety valve	3	20	makeup water pump	2
3	primary coolant pump	4	12	chimney	1 or 2	21	boric acid storage tank	1
4	pressurizer	1	13	emergency air cooler	2	22	boric acid makeup tank	1
5	relief tank	1	14	water storage tank	1	23	spent fuel pit cooling pump	2
6	core makeup tank	2	15	regenerative heat exchanger	1	24	spent fuel pit heat exchanger	2
7	accumulator	2	16	letdown heat exchanger	1	25	spent fuel pit	1
8	emergency water tank	2	17	mixed bed exchanger	1	26	protective shell	1
9	special sump	2	18	mixed bed exchanger	1	27	containment (steel)	1

FIG. 5.1-1. AC-600 nuclear island, flow diagram

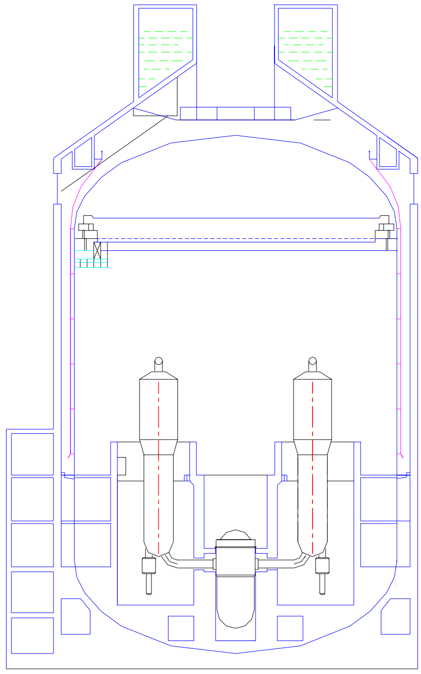


FIG. 5.1-2. Reactor building arrangement

5.1.2.3 Fuel handling and transfer systems

No information provided.

5.1.2.4 Primary components

Reactor pressure vessel

The reactor vessel (Figure 5.1-3) encloses all components of the reactor core. It is made of SC 508-3 steel made in China. Due to the lower power density in the core and the larger vessel inside diameter, it is considered to be much safer during the 60 years design life of the plant. The nozzles of the control rod drive mechanisms (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.

Reactor internals

The reactor internals of AC-600 consist of upper internals, lower internals, hold-down spring assembly and control rod guide tube assembly. This component is based on the proven design which has been used in Qinshan phase II and several improvements are described below:

Upper support columns serve as guide columns for incore neutron flux detectors.

Radial neutron reflector is adopted to replace baffle and heat shielding assembly.

CRDM

This component is based on the proven design which has been adopted in Qinshan Phase Ⅱ and several improvements are as below:

Extend service life, to satisfy the need of 60 years NPP design life.

Adopt electromagnetic coil which is resistant to high temperture, so the operation temperature of the CRDM is higher than 300℃ (about 350℃).

Intergrated head package

An itergrated head package will be adopted in the design of AC-600 to decrease reactor shutdown time and personnel dose as well as save space.

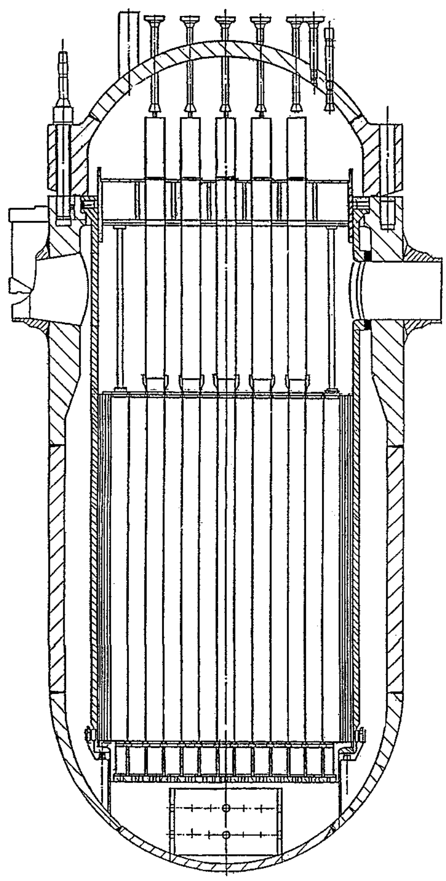


FIG. 5.1-3. *Reactor vessel and internals*

Steam generators

The steam generators are improved on the base of 60F used in Qinshan phase II. The material of the U-tubes is Inconel-690 TT. Two canned pumps are welded reversely on the steam generator bottom head. In this case, the U shape cross-over leg of the coolant pipe is eliminated.

Pressurizer

The pressurizer is larger than that of the Qinshan phase II. Its total volume is 40 m^3.

Reactor coolant pumps

The reactor coolant pumps (RCPs) are of the mixed flow, canned motor pump type. There are four canned pumps connected to the steam generator bottom heads directly. Lubrication and cooling of the RCPs are performed with water. To increase the rotating inertia of the canned motor pump, a motor and pump design with a rotating inertia of 0.20 -m^2 will be employed.

Main coolant lines

The inside diameter of the hot and cold leg nozzles is 787.4 and 559 mm, respectively.

5.1.2.5 Reactor auxiliary systems

5.1.2.5.1 Component cooling water system

The component cooling water system transfer heat from various nuclear island components that need to be cooled to the service water system during normal phase of operation. Also it provides a barrier between components being cooled that handle radioactive fluid and the service water system , so the barrier can prevent the release of radioactivity resulted from the damage of radioactive fluid bound to the environment.

The component cooling water system is non-safety-related, closed loop cooling system. It consists of two component cooling water pumps, two component cooling water heat exchangers, one component cooling water surge tank and associated valves, piping and instrumentation. The system components are arranged into two trains. Each train consists of one component cooling water pump and one component cooling water heat exchanger. The component cooling water surge tank connected to the intake of the component cooling water pumps accommodates thermal expansion and contraction of the primary system volume, and it also provides pressure head of the pumps.

5.1.2.5.2 Spent fuel pit cooling and purification system

The spent fuel pit cooling and purification system performs the function of removing decay heat generated by stored fuel assemblies from the water in the spent fuel pit. It also performs the function of purification of the water in the spent fuel pit and the refueling tank.

The spent fuel pit cooling and purification system is non-safety-related system. It consists of spent fuel pit cooling circuit and spent fuel pit surface skimming circuit. The spent fuel pit cooling circuit is divided into two trains. Each train consists of one spent fuel pit pump, one spent pit heat exchanger, one spent pit demineralizer and one spent pit filter. Spent fuel pit surface skimming circuit consists of one skimming pump and one skimming filter.

5.1.2.5.3 Chemical and volume control system

Chemical and volume control system performs the functions of purification, inventory control, water chemical control, oxygen control in the reactor coolant system as well as borate concentration control in nuclear auxiliary system.

Chemical and volume control system is non-safety-related system. It consists of one regenerative heat exchanger, one letdown heat exchanger, three demineralizers (two mixed bed demineralizers and one cation bed demineralizer), three filters (one makeup filter and two purification filters), one boric acid tank, one boric acid batching tank, one chemical mixing tank and associated valves piping and instrumentation.

5.1.2.6 Operating characteristics

No information provided.

5.1.3 Description of turbine generator plant system

5.1.3.1 Turbine generator plant

No information provided.

5.1.3.2 Condensate and feedwater systems

No information provided.

5.1.3.3 Auxiliary systems

No information provided.

5.1.4 Instrumentation and Control Systems

5.1.4.1 Design concepts, including control room

The AC600 Instrumentation and Control (I&C) Systems, which provide nuclear power plant with various monitoring and control means, combine with controlled processes and operators to ensure the plant's safe, reliable and economical operation. This I&C system design is based on advanced and proven computer control, network communication and human factor engineering technology and backed up minimized analogue devices.

AC600 I&C systems have a four-level structure that is divided into input/output level, automation level, operation & display level and plant management level. The whole I&C systems consist of man-machine interface system, automation control and protection system and communication system.

Man-machine interface system orientates to operator and includes data display & process system and operation & control center system.

Control room belongs to operation & control center system. In the main control room, Man-machine interface devices include large screen, redundant operator stations and auxiliary control panel etc. These device and relevant Man-machine resources are used for monitoring, assessing and diagnosing all operational conditions of nuclear power plant and its equipment. They also provide operator with timely alarm and processed information against events and accident conditions, present post-accident monitoring information and assist operation staff with powerful means.

Auxiliary control panel adopts hard-wire technology and acts as a kind of back-up device in case of computer system failure.

If the main control room is unavailable, the emergency shutdown control room is used for realizing to shutdown nuclear power plant to cooldown condition.

Automatic control and protection system orientates to plant. It is made up of the following systems:

- Reactor control system;
- Reactor protection system;
- Post-accident monitoring system;
- Plant control system;
- Dedicated monitoring systems;
- Hard-wire manual system etc.

Communication system includes high-speed and redundant industrial local area networks. It connects with man-machine interface system and automation control & protection system. It makes I&C systems an integrated system.

During normal operation, reactor power and process parameters are maintained automatically by reactor control system in order to prevent unsafe condition.

The design value of the reactor neutron flux is maintained within ±2% of its nominal value by means of the control bank of neutron absorbers, consisting of several rod cluster control assemblies.

The design value of the primary pressure is maintained within ±0.3 MPa by means of 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.

The design value of the secondary pressure is maintained within ±0.2 MPa by an appropriate balance of reactor power and steam flow from the steam generators to the turbine or to the steam dumping devices.

The design value of the water level in the steam generators is maintained within ±180 mm of its nominal level by means of the steam generator feedwater controller, actuating the control valve on the steam generator feedwater line.

The design value of the water level in the pressurizer is maintained by the level controller, actuating the control valves located on the make-up line, and make-up pumps.

Reactor control system includes the following subsystems:

- Reactor power control system;
- Pressurizer pressure control system;
- Pressurizer level control system;
- Steam generator pressure control system;
- Steam dump control system;
- Feedwater pump speed control system;
- Motor valve control system etc.

In case events or accident conditions happen, reactor protection system automatically initiates various protection actions to limit the evolution of accidents and mitigate the consequences of events, and also it provide control room with relevant information including alarm signals.

Based on optimized AC600 I&C conceptual design, the preliminary prototype equipment of reactor control system, reactor protection system and a part of systems in advanced control room are developed in laboratory.

The research content of reactor control system includes the selection of NPP control strategy, advanced Distributed Control system (DCS)product application and high reliable design of function and structure in control system etc.

In development of prototype equipment of reactor protection system, the representative protection parameters are selected; Four protection instrument channels and two logic protection trains are adopted; High reliable industrial control computerized components are used; embed-in, real-time and multi-task operating system software and determinative communication networks are applied. During research process, safety software verification and validation actives are performed.

The research content of advanced control room system involves man factor engineering, plant data communication, operation and control system, intelligent alarm system, operator support system, design of information display

5.1.4.2 Reactor protection and other safety systems

The degree of automation is such that reactor safety will be ensured fully by automatic control and protection systems during the first 30 minutes after an accident has happened. In design basis accident situations, interventions by the operators should not take place within the first 30 minutes in order to allow the operators enough time to consider the characteristics of the accident, which is seen as an effective countermeasure to prevent erroneous actions. The reactor will be "walk-away" safe during this period.

In addition, in design basis accident situations provisions are made for:

- 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 etc.

The above information is provided in large screen, operator stations and auxiliary control panel. During accidents, the devices for automatic event recording can also be used. The post-accident monitoring system is designed to collect and to present the important post-accident plant data, such as the information on the core thermoelectric couple system, the core coolant and pressure vessel level monitoring system and some important plant variables.

Safety systems encompass:

- 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 for the containment;
- Secondary overpressure protection system;
- Diesel-generator sets system and
- System of reliable direct current power supply.

Except for the two last systems, these systems are described in Section 5.1.6.2. The remaining two items are discussed in Section 5.1.5.2.

The reactor protection system is designed to select necessary and sufficient protection parameters. Once the protective set points are exceeded, it will automatically initiate protection actions to ensure the safety of reactor, the devices and the personnel of nuclear power plant, to prevent the release of radioactive material to surrounding environment. It consists of reactor trip system and Engineering Safeguard Feature Actuation System (ESFAS).

The function of reactor trip system is to end the unsafe condition. The function of ESFAS is to mitigate the unsafe condition. Furthermore, in each redundant protection channel, two independent and separate protective-function groups are provided to ensure diverse protective ability against the same accident.

The protection parameters of AC600 reactor protection system are the following:

- High of neutron flux;
- High of neutron flux rate;
- High LPD and low DNBR;
- High of pressure in pressurizer;
- Low of pressure in pressurizer;
- High of lever in pressurizer;
- Low of the flow rate of the reactor coolant;
- High of water level in SG;
- Low of water level in SG;
- Low of the reactor coolant pump speed;
- High of containment pressure;
- Signal of low-pressure safety injection;
- Low of steam line pressure;
- High of steam flow rate etc.

Note: In data processing and calculating part of reactor protection system, LPD (Linear Power Density) is calculated by inputting reactor power level, control rod position and current of 6 section ionization chambers which belongs to 4 power ranges; DNBR (Departure from Nucleate Boiling Ratio) is calculated by inputting pressurizer pressure, power level, reactor coolant flow rate, reactor inlet temperature, radial nuclear enthalpy rise factors and axial unified power distribution. As one of protection function groups, LPD, DNBR and a few of other protection parameters are incorporated in one computer unit.

When the protection parameters are beyond pre-set limits, the reactor trip system will scram trip circuit breakers, which de-energize the control rod drive mechanisms, so that the control rods will drop down the reactor core. In this case, the disappearance of initial actuation signal will not stop the reactor trip action.

If protective actions are needed according to accident condition, ESFAS will send safeguard signals to actuate the ESF systems in order to alleviate or mitigate the consequences of design base accidents or beyond design base accidents. For the philosophy of passive design of safety system is adopted, the design of output signals of ESFAS is simplified.

In case of reactor protection system failure, ATWT (Anticipated Transients Without Trip) system is designed. It will initiate some protective actions such as turbine trip and the start of auxiliary feedwater and also reactor trip, in order to make the consequences be acceptable.

5.1.5 Electrical systems

5.1.5.1 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.

5.1.5.2 Safety-related systems

Diesel generator sets system

In case of the failure of main grid and auxiliary grid connecting with two diesel generator sets, this system will be put into operation. The diesel generators will reach a rated speed and a rated voltage within 10 sec after they receive the signal of start-up command.

The two diesel generators sets provide power supply to 2-train buses. They are located in 2 separate rooms.

System of reliable direct current power supply

This system consists of storage batteries, relevant charge devices and distribution facilities. It provides power supply to the operational equipment of safety systems and to the electromagnetic circuits for recording of necessary post-accident parameters.

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 storage batteries during a station blackout situation is provided for main control room and emergency shutdown control room in full capacity.

5.1.6 Safety concept

5.1.6.1 Safety requirements and design philosophy

The basic safety requirements can be summarized as:

- During normal conditions and operational occurrences, radioactive material shall be confined by maintaining the integrity of all physical barriers in the defence-in-depth: fuel rod claddings, primary pressure boundary and steel containment.

- Radioactive materials released from the primary circuit in design basis accidents shall be confined by maintaining steel containment integrity.

- In the event of design basis accidents with a leak from primary to the secondary circuit the release of radioactive material shall be controlled by isolation of the steam generators on the steam and water side by means of quick-acting shut off isolation valves, actuated by a signal of radioactivity increase in the damaged steam generator.

- Radioactive materials released from the fuel and the primary circuit in beyond-design accidents shall be confined by the steel containment and by operation, if necessary, of the filtration plant for controlled removal of the atmosphere inside the containment.

In order to meet these requirements, the AC-600 design is based on utilization of a combination of active and passive features. Quite obviously, controls and countermeasures related to normal operation and occurrences rely on active systems and components in the same way as currently operating PWRs, but when it comes to accident situations, the strategy involves using passive safety features, as far as possible and practical for short term actions and responses, whereas active systems are provided as back-up and means for corrective and mitigative actions in the longer term.

The essential safety functions in emergency situations are to shut the reactor down, and to remove the residual heat. Shutdown is normally accomplished by a reliable reactor trip system, in case of ATWT, boron water of the safety injection system can be actuated by operators. The safety injection system which serves to provide make-up water in the event of leakages, comprises a set of three systems, the two systems for high and medium pressure range injection are passive, whereas the third, for low pressure injection, is an active system. Residual heat is removed by means of completely passive system functions in the event of station blackout.

In the first stage of a 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 then generated is through the turbine bypass system conveyed to a condenser to be condensed. The auxiliary feedwater system supplies the steam generator with water. The whole process goes on till the pressure of the coolant system has decreased to 2.8 MPa and the temperature to 180℃.

In the second stage of the 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 a heat exchanger cooled by equipment coolant. During normal plant operation, this system serves as the spent fuel pool cooling and purification system. During plant shutdown, one of its trains is used for reactor residual heat removal. At the same time, the spent fuel pool is also cooled till the coolant pressure is below 0.1 MPa. Coolant temperature decreases to and remains at cold shutdown temperature.

During the plant shutdown and cooling process, the coolant pumps are always in operation; they do not stop until the coolant temperature has decreased to 70℃. During this period, reactor coolant is circulated through the coolant loops. After that, the coolant is circulated by the spent fuel pit cooling pump.

In the event of a large leakage from the reactor coolant pressure boundary, the residual heat is removed by the safety injection or emergency core cooling system and the passive containment cooling system. Make-up water to the primary system to ensure that the fuel assemblies in the pressure vessel will remain covered with water is provided by the emergency core cooling system.

As long as the reactor coolant pressure boundary is completely intact, a station blackout is a typical situation in which emergency residual heat removal will be required. In such case, 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 the atmosphere. By this system, the coolant temperature and pressure can be kept within acceptable limits until the power supply has been restored, or brought down to values corresponding to cold shutdown. 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.

Deterministic design basis

No information provided.

5.1.6.2 Safety systems and features (active, passive and inherent)

The major safety systems with respect to shutdown, core cooling, and overpressure protection are:

- Primary circuit overpressure protection system,
- Emergency core cooling system,
- System of passive residual heat removal from the secondary side of the steam generator,
- Passive cooling system for the containment,
- Secondary circuit overpressure protection system,
- Auto depressurize system,
- Containment combustible gas control system,
- main control-room habitability system.

which are described further in the following.

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.

Emergency core cooling system

The emergency core cooling system (ECCS) comprises the following complex subsystems initiated automatically:

- System of core make-up tank with full pressure (high pressure safety injection subsystem);
- System of accumulator with nitrogen under pressure;
- System of low pressure active safety injection and recirculation.

In order to increase the reliability of the safety injection, two full pressure core makeup tanks, two accumulators and two low head safety injection/recirculation pumps, which are installed in the containment sumps, are utilized in the AC-600 design. AC power supplies are not needed for fulfillment of the ECCS functions, except for the subsystems of low pressure active safety injection and recirculation.

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 a conventional design. In order to ensure the functions of the low head safety injection/recirculation system, utilization of active pumps has been found necessary. 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).

System of passive residual heat removal from the secondary side of the steam generator

The passive emergency residual heat removal system is mainly used to remove the residual reactor power in the event of a station blackout by natural circulation in the primary coolant system and on the secondary side of the steam generator. It may also be effective in a main steam line rupture or loss of feedwater event. The system consists of two independent trains, each of them being connected to the reactor coolant loops via the respective steam generator. Each train has an emergency feedwater tank, an emergency air cooler located_in a chimney outside the containment, and piping (and valves) for steam and condensate circulation. The fail-open pneumatic-isolation valves on the piping are driven by compressed air. The air-cooler with the help of chimney rejects the core decay heat transferred from the steam generators into the atmosphere

Passive cooling system for the containment

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. The steam released is condensed on the inside of the containment shell which is cooled on the outside by a natural circulation air flow and a gravity-driven water spray onto the steel shell surface by water from the elevated tanks on top of the containment. The heat released inside the containment is rejected to the atmosphere from the containment, and the pressure and temperature inside the containment decrease. The pressure of the atmosphere inside the containment is kept below the permissible design value.

A specific computer code PCCSAC-3D for containment key parameter caculation (such as peak pressure and temperture) has been developed. The caculation result demonstrates that the design of AC-600 containment meets the design limits.

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.

Automatic depressurization system

This system consists of two redundant groups of depressurizaiton valves, a relief pipe and a relief tank. The relief pipe and relief tank are shared with RCS.

ADS is automatically actuated by the low water-level of core makeup tank.,which reduces the pressure of RCS and ensures that the medium head safety injection subsystem and low head safety injection subsystem can be initiated in time.

Containment Combustible control system

This system is used to monitor and to keep the hydrogen concentration in containment within regulated limit after DBAs or severe accidents, to prevent the potential risk of explosion caused by the combination of hydrogen and oxygen, and then to ensure the integrity and leaktightness of containment.

This system consists of three subsystems as below:

- Hydrogen concerntration monitor subsytem mainly comprises some hydrogen -detectors distributed in containment;
- Hydrogen recombination subsystem mianly comprises two hydrogen recombination systems;
- Hydrogen ignition subsystem mainly comprises some hydrogen ignitors.

Main control-room habitability system

It is a passive system which comprises compress air tanks, pressure control valves, isolation valves etc. When the HAVC of NI is unavailable or high radioactive signal is detected in the air of main control-room, this system will actuate automatically, providing the main control-room with fresh air and pressurization to maintain the situable habitable condition .

Application of passive safety system in LOCA

The analysis and study of LOCA demonstrates :

In the case of LOCA, the ECCS can inject water into the core in time,and retain the water inventory of RCS as much as possible. During the whole process, this system can maintain core cooling and prevent the core from damage.

During large LOCA or MSLB condition, PCCS will actuate automatically, and remove heat from containment as much as possible to prevent the containment from damage.

ADS mainly contributes to the medium or small LOCA, and it can effectivly reduce the pressure of RCS to ensure the appropriate actuation of safety injection system and core cooling.

So in the case of LOCA, the passive safety systems of AC-600 can fufill their functional requirment and ensure the safety of the reactor.

5.1.6.3 Severe accidents

Accident prevention and mitigation, including severe accident, has been a necessary component of AC-600 design activity. One important policy in the design is to prevent the accident from progressing to core damage. In case core damage begins, the features to maintain the integrity of the fission products boundary are also considered in the design.

The design of AC-600 includes some special advanced design features to prevent core damage. The larger pressurizer and lower power density, as compared to operating plant, increase transient operations margins and safety margins. The passive safety related systems rely on natural forces, such as gravity and stored energy, to perform their safety functions. Several important contributions to risk for operating nuclear power plant have been eliminated. The function of automatic depressurization system (ADS) is to provide a safety-related means of reducing RCS pressure in a controlled fashion during accidents to allow safety injection. This constitutes the bleed portion of the feed-and-bleed means of core cooling. The AC-600 control room design using digital I&C systems is an advanced design that is expected to provide more as well as more useful information to the operator during an accident.

The AC-600 design also includes design features to mitigate core damage consequence. The ADS can reduce or eliminate the potential for creep rupture of steam generator tubes and the reactor vessel. Prevention of reactor vessel breach precludes severe accident phenomena associated with vessel failure -direct containment heating (DCH), large hydrogen combustion, ex-vessel steam explosion, and core concrete interactions- thereby reducing the probability of early containment failure. Hydrogen control system composed of hydrogen combiners and igniters can effectively limit the hydrogen accumulation, eliminate the threat to the integrity of containment. The intent of igniters is to ignite the hydrogen as soon as sufficient hydrogen has accumulated to achieve a combustible mixture. This early combustion is intended to limit the hydrogen concentration well below the 10-percent limit. The hydrogen ignition system can be working for 72 hours. The steel containment, which relies on air cooling of the passive containment cooling system (PCCS), has the capability of effectively mitigating the core melting without time limitation.

In addition, an improved emergency operation procedure (EOP) will be used to best use the systems available for the purpose of preventing core damage, and severe accident management guidelines (SAMGs) for the purpose of halting the progression of a severe accident by protection of the primary system boundary, the containment.

5.1.7 Plant layout

5.1.7.1 Buildings and structures

A typical site plan for the single unit AC600 has been designed. It meets the requirements of earthquake, hydrological geology, engineering geology, meteorology, fire fighting, transportation, radiation protection, flood protection, security guard, etc.

The power generation complex consists of 7 principal building structrues; the reactor building, the turbine generator building, the fuel building, the electrical building, the auxiliary building, the diesel generator building and the radwaste building. The reactor building, the fuel building, the electrical building and the auxiliary building structures are constructured on a common basemat. AC600 site plan will be tailored to the specific topography and characteristic of the selected constructed site.

5.1.7.2 Nuclear island building design and arrangement

AC600 nuclear island buildings consist of the reactor building, the fuel building, the electrical building and the auxiliary building. They are constructed on a common reinforced concrete foundation.

The reactor building consists of the shielding building and the containment building, both of which are constructed on a common basemat.

In the nuclear island arrangement, fire areas have been divided according to physical separation principle. The plant is divided into fire areas and fire zones to isolate potential fires and minimize the risk of spread of fire. The same safety-related trains are located in different fire areas or different fire zones so as to protect the safety systems from the potential common mode failure.

To meet ALARA requirements all nuclear island rooms and radiation zones have been classified according to radioactive dose level. The radioactive control areas are divided into 4-type zones; red zone, orange zone, yellow zone and green zone.

Design bases

The design of the AC-600 is taking into account two levels of seismic events: an operation basis earthquake (OBE) of magnitude 7 on the MSK-64 scale and a safe shutdown earthquake (SSE) of magnitude 8 on the MSK-64 scale. The peak ground acceleration used in design is 0.2g.

The seismic effects for reactor plant equipment are calculated. During the operating basis earthquake, normal operation of the reactor plant is to be maintained. During the safe shutdown earthquake, safe plant shutdown and adequate cooling of the core must be ensured. All civil structures, process components and equipment, pipelines, instrumentations, and so on, depending upon the degree of their responsibility for safety ensurance during seismic events and availability after an earthquake, are divided into 3 seismic categories. Components and systems of category 1 shall fulfill their safety functions during and after an earthquake of SSE intensity. After an OBE, availability is maintained.

The seismic category 1 includes:

- Systems for normal operation, failure of which during an SSE may result in radioactivity releases causing excessive population doses in comparison with the specified values for SSE condition;
- Safety systems for keeping the reactor in a sub-critical state, for emergency heat removal and for confinement of radioactive products;
- Structures and equipment which could impair these functions as a consequence of an SSE.

The designers considered the possibility of using special seismic isolators located under the base plate to minimize the seismic impact on structures and equipment.

The external wind load for the first category buildings and structures is assumed to arise from a hurricane with wind speed of 25 m/s.

With respect to tornado effects, the following characteristics and physical parameters are taken into account in the design for the first category buildings and constructions:

- Maximum horizontal speed of rotation of tornado wall: 85 m/s;
- Translational motion speed of tornado: 22 m/s;
- Tornado radius: 45 m;
- Differential pressure between center and periphery of the whirlwind: 8.5 kPa;
- Impact of missiles carried away by a whirlwind with a speed of 26 m/s.

With respect to external explosions and airplane crashes, the following characteristics are specified:

- Front pressure of explosion shock wave: 50kPa;
- Duration of compression phase: 300 ms;
- Direction of propagation is horizontal;
- Impact of a plane with 5.7t mass at a speed of 100 m/s.

Shield building

The AC600 shield building is the structure and annulus area that surrounds the containment building. An air baffle, located between the steel containment vessel and the concrete shield building, which defines the cooling airflow path.

During normal operations, a primary function of the shield building is to provide shielding for the containment vessel and the radioactive sources located in the containment building.

Another function of the shield building is to protect the containment building from external events. The shield building protects the containment vessel and the reactor coolant system from the effects of tornadoes and tornado produced missiles.

The shield building is a seismic category I reinforced concret structure with a conical roof. It is also an integral part of the passive containment cooling system(PCCS). A PCCS water storage tank, air inlets, an air difusser, a PCS valve room, a person basket are incorporated into the shield building and its conical roof. Wire meshes are located at the air inlets and the air outlet to protect the containment vessel from the external missles. The PCS water storage tank has a stainless steel liner that provides a leakage barrier on the inside suface of the tank.

Modularization technology has been taken into account in the design of the shielding building and the in-containment structures so that they can be constructed in modules and reduce the period of construction.

5.1.5.3 Containment

The containment, a seismic category I structure, is a freestanding cylindrical steel containment vessel with elliptical upper and lower heads. Its lower head is embedded in the reinforced concret foundation. A water flow distributor is mounted on the outside surface of the upper head which can ditribute the water dropped from the PCS water storage tank into symmetrical water film on the outside surface of the containment vessel.

The steel containment is designed to withstand an internal overpressure of 0.4 MPa and a maximum temperature of 134℃ with a very high leaktightness; the leakage shall not exceed 0.3% of volume per day. During design basis accidents, the containment systems ensure confinement of radioactive material inside the steel containment, heat removal from the hermetic steel containment, and control and suppression of hydrogen.The wall thickness has been calculated according to the peak value pressure of the in-containment atmosphere in the event of postulated accidents. The size and energy of missiles inside the steel containment are determined in the design based on the "leak before break" concept. The mechanical effect of missiles and steam-water jets on the steel containment is excluded by means of a protective shield.

The wall thickness increases in the lower course of the cylindrical shell to provide margins for protection in the event of corrosion in the embedded transition region. The cylindrical portion of the containment vessel is provided stiffeners so as to maintain the integrity of the containment in the event of designed basis accidents.

Following LOCA, the containment hydrogen control system is provided to limit the hydrogen concentration in the containment so that containment integrity is not endangered.

Modularization technology has been taken into account in the design of the containment vessel so that it can be constructed in modules.

Containment ventilation system

The containment ventilation system consists of the containment recirculation cooling subsystem, the reactor cavity cooling subsystem, and the containment air filtration subsystem.

The containment recirculation cooling subsystem is a fan coil unit subsystem which controls the containment air temperature and humidity to provide a suitabe enviroment for equipment operability and personnel accessibility during normal operation, refueling and plant shutdown.

The reactor cavity cooling subsystem is designed to provide control of the reactor cavity area concrete temperature during normal operation.

The containment air filtration subsystem purges the containment atmosphere of airborne radioactivity for personnel access. The subsystem is operated intermittently during normal plant operation, containment preaccess periods, reactor shutdown, and when high airborne radioactivity is detected in the containment or in the exhaust air. The subsystem is not required to be operational after a design basis accident.

5.1.7.4 Turbine generator building

The turbine generator building encloses and supports the main turbine, generator and associated fluid and electrical systems. This building typically encloses the following major equipment: TG, TG pedestal, condenser, water chillers, condensate demineralizer system equipment, low and high pressure heaters, deoxidize equipment, etc. It provides weather protection for the laydown and maintenance of major turbine/generator components.

The turbine generator building is a steel column and beam structure which has a length of about 100m and a width of about 45m. The turbine building ground floor is a reinforced concret slab.

5.1.8 Technical data

General plant data		
Power plant output, rated	600	MWe
Power plant output, net		MWe
Reactor thermal output, rated	1930	MWt
Reactor thermal output, NSSS	1936	MWt
Power plant efficiency, net		%
Nuclear steam supply system		
Number of coolant loops	2	
Primary circuit volume, including pressuriser		m^3
Steam flow rate at nominal conditions [1951t/h]	2168	kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature, at outlet nozzle	282.94	^oC
Steam pressure, at upstream of the restrictor (abs)	6.71	Mpa
Steam moisture, at outlet nozzle		<0.1%
Feedwater temperature	230	^oC
Reactor coolant system		
Primary coolant thermal design flow rate/loop	23750	m^3/h
Primary coolant nominal flow rate/loop	24740	m^3/h
Primary coolant mechanical design flow rate/loop	25730	m^3/h
Reactor operating pressure, absolute pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet,	293	^oC
Coolant outlet temperature, at RPV outlet	327	^oC
Mean temperature rise across core		^oC
Reactor core		
Active core height	3.658	m
Equivalent core diameter	2.92	m
Heat transfer surface in the core	4179	m^2
Total fuel inventory [UO_2]	66.8	t
Average linear heat rate	13.426	kW/m
Average fuel power density	32.9	kW/kg U
Average core power density (volumetric)	78.69	kW/l
Thermal heat flux	450	kW/m^2
Fuel material	Sintered UO_2	
Fuel assembly total length	~4100	mm
Rod array	square, 17×17 -25 (AFA-3G)	
Number of fuel assemblies	145	
Number of fuel rods/assembly	264	
Number of guide tubes for control rods/instr.	24/1	
Number of spacers	11	
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 batch discharge burnup of fuel	⩾45 000	MWd/t
Cladding tube material	M5(Zr-Nb-O alloy)	
Cladding tube wall thickness	0.57 or 0.64	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly	667.8	kg
Active length of fuel rods	3657.6	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	mm/min (or mm/s)	
Soluble neutron absorber		
Reactor pressure vessel		
Cylindrical shell inner diameter	4 000	mm
Wall thickness of cylindrical shell	205	mm
Total height (to the upper surface of CRDM seat)	12395mm	
Base material: cylindrical shell	low alloy steel SC508-3	
RPV heads	low alloy steel SC508-3	
Design pressure/temperature	17.2/343	$MPa/^oC$
Total net weight	~450	t

Steam generators

Type	Advanced 60F	
Number	2	
Heat transfer surface	5 630	m^2
Number of heat exchanger tubes	4 640	
Tube dimensions	19.05x1.09	mm
Maximum outer diameter	4488	mm
Total height	~21375	mm
Total net weight	~340	t
Shell and tube sheet material	SC508-III	
Tube material	Inconel 690-TT	

Reactor coolant pump

Type	Single-stage centrifugal pump with canned motor	
Number	4	
Design pressure/temperature	17.2/343	MPa/°C
Nominal flow rate	12370	m^3/h
Pump head	~60mH_2O	
Rotation inertia	210 kg.m^2	
Rated power of motor	3000	kW
Pump casing material	Stainless steel	
Pump speed	1 500	rpm

Pressuriser

Total volume	40	m^3
Water/steam volume, in rated condition	25/15	m^3
Design pressure/temperature	17.2/360	MPa/°C
Heating power of the heater rods	1 440	kW
Number of heater rods	60	
Total height	12846	mm
Material	SC508-3	
Total net weight	~80	t

Pressuriser relief tank

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Dry, double wall, in steel/concrete	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	38/63.38	m
Free volume	75 000	m^3
Design leakage rate	< 0.25	vol%/day
Is secondary containment provided?	yes, space between the walls	

Reactor auxiliary systems

Reactor water cleanup, capacity		kg/s
filter type		
Residual heat removal, at high pressure		kg/s
at low pressure		kg/s
Coolant injection, at high pressure		kg/s
at low pressure		kg/s

Power supply systems

Main transformer, rated voltage		kV
rated capacity		MVA
Plant transformers, rated voltage		kV
rated capacity		MVA
Start-up transformer 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	MW
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

<u>Turbine plant</u>

Number of turbines per reactor	1	
Type of turbine(s)		
Number of turbine sections per unit (e.g. HP/LP/LP)		
Turbine speed		rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/°C

<u>Generator</u>

Type	
Rated power	MVA
Active power	MW
Voltage	kV
Frequency	Hz
Total generator mass	t
Overall length of generator	m

<u>Condenser</u>

Type	
Number of tubes	
Heat transfer area	m^2
Cooling water flow rate	m^3/s
Cooling water temperature	°C
Condenser pressure	hPa

<u>Condensate pumps</u>

Number	
Flow rate	kg/s
Pump head	
Temperature	°C
Pump speed	rpm

<u>Condensate clean-up system</u>

Full flow/part flow	
Filter type	

<u>Feedwater tank</u>

Volume	m^3
Pressure/temperature	MPa/ °C

<u>Feedwater pumps</u>

Number	
Flow rate	kg/s
Pump head	
Feedwater temperature	°C
Pump speed	rpm

<u>Condensate and feedwater heaters</u>

Number of heating stages	
Redundancies	

5.1.9 Measures to enhance economy and maintainability

a) Design life increase from 40 years to 60 years.

b) System simplification, adoption of the passive safety systems and separation of the safety systems with normal operation systems.

- Reduction of safety class 2 and class 3 pumps, valves and piping;
- Reduction of anti-seismic buildings;
- Reduction of system operation and maintenance.

c) Use of advanced core design and low leakage Fuel Management, such as,

- Increase of thermal neutron usage;
- Fuel reload cycle increased from 12 months to 18-24 months, compared with QS-II NPP;
- Decrease of the critical Boron Concentration, hence promoting operation flexibility, and reducing the waste production.

d) Use of advanced technologies, mainly,

- Use of LBB, cancellation of the whip restraints, reduction of the piping snubbers;
- System simplification, reduction of the pumps, valves and piping;
- use of the RV Integrated Head Package, decrease the refuelling time;
- Cancellation of the temperature-measuring bypass, and therefore reducing the maintenance work and personnel exposure.

Due to the above mentioned measures, the capital investment and maintenance work amount are reduced.

e) Use of advanced technologies, such as modularity, to effectively shorten the construction period, compared with QS-II, the construction period, reduced from 68 months to possibly 54 months.

As the detailed study has not been performed, the comparison between AC600 and QS-II NPP could not been given.

5.1.10 Project status and planned schedule

The main R&D results will be applied to the development of large-scale advanced PWR, the proposal of which has been submitted to governmental organizations, and will hopefully be approved soon.

5.2 AP-600 (WESTINGHOUSE, USA)

5.2.1 Introduction

The Westinghouse Advanced Passive PWR AP-600 is a 600 MW(e) pressurized water reactor (PWR) with advanced passive safety systems and extensive plant simplifications to enhance the construction, operation, and maintenance of the plant. The plant design utilizes proven technology which builds on approximately 40 years of operating PWR experience. PWRs represent 74 percent of all Light Water Reactors around the world, and the majority of these are based on Westinghouse PWR technology.

The AP-600 is designed to achieve a high safety and performance record. Safety systems maximize the use of natural driving forces such as pressurized gas, gravity flow and natural circulation flow. Safety systems do not use active components (such as pumps, fans or diesel generators) and are designed to function without safety-grade support systems (such as AC power, component cooling water, service water, HVAC). The number and complexity of operator actions required to control the safety systems are minimized; the approach is to eliminate a required operator action rather than to automate it. The net result is a design with significantly reduced complexity and improved operability.

The AP-600 standard design complies with all applicable US NRC criteria. Extensive safety analysis has been completed and documented in the Standard Safety Analysis Report (SSAR) and Probabilistic Risk Analysis (PRA) submittals to the NRC. An extensive testing programme has been completed, and verifies that the plant's innovative features will perform as designed and analyzed. PRA results show a very low core damage frequency which meets the goals established for advanced reactor designs and a low frequency of release due to improved containment isolation and cooling. The AP-600 design received Final Design Approval from the US NRC in September 1998 and Design Certification in December 1999.

An important aspect of the AP-600 design philosophy focuses on plant operability and maintainability. These factors have been incorporated into the design process.

The AP-600 design includes features such as simplified system design to improve operability while reducing the number of components and associated maintenance requirements. In particular, simplified safety systems reduce surveillance requirements by enabling significantly simplified technical specifications.

Selection of proven components has been emphasized to ensure a high degree of reliability with a low maintenance requirement. Component standardization reduces spare parts, minimizes maintenance training requirements, and allows shorter maintenance durations. Built-in testing capability is provided for critical components.

Plant layout ensures adequate access for inspection and maintenance. Laydown space for staging of equipment and personnel, equipment removal paths, and space to accommodate remotely operated service equipment and mobile units have been considered as part of the plant design. Access platforms and lifting devices are provided at key locations, as are service provisions such as electrical power, demineralized water, breathing and service air, ventilation and lighting.

The AP-600 design also incorporates radiation exposure reduction principles to keep worker dose as low as reasonably achievable (ALARA). Exposure length, distance, shielding and source reduction are fundamental criteria that are incorporated into the design.

Various features have been incorporated in the design to minimize construction time and total cost by eliminating components and reducing bulk quantities and building volumes. Some of these features include the following:

- The flat, nuclear island common basemat design effectively minimizes construction cost and schedule;
- Utilization of the integrated protection system, the advanced control room, distributed logic cabinets, multiplexing, and fiber optics, significantly reduces the quantity of cables, cable trays, and conduits.
- The stacked arrangement of the Class 1E battery rooms, the dc switchgear rooms, the integrated protection system rooms, and the main control room. This stacked arrangement eliminates the need for the upper and lower cable spreading rooms that are required in the current generation of PWR plants.
- Application of the passive safety systems replaces and/or eliminates many of the conventional mechanical safety systems that are typically located in the Seismic Category I buildings in current PWR plants.

The AP-600 is designed with environmental consideration as a priority. The safety of the public, the power plant workers, and the impact to the environment have all been addressed as specific design goals, as follows:

- Operational releases have been minimized by design features;
- Aggressive goals for worker radiation exposure have been set and satisfied;
- Total radwaste volumes have been minimized;
- Other hazardous waste (non-radioactive) have been minimized.

The AP-600 Nuclear Power Plant was designed by Westinghouse under the sponsorship of the US Department of Energy (DOE) and the Electric Power Research Institute (EPRI). The design team included a number of US and foreign companies and organizations, such as Bechtel, Burns & Roe, Initec (Spain), UTE (Spain), and Ansaldo (Italy) as architect engineers, Avondale Industries (module design), CBI Services, Inc. (containment vessel design), M-K Ferguson Co. (constructability, schedule, and cost estimation), Southern Electric International (turbine island buildings and systems), ENEA Energy Research Center of Italy (tests of the automatic depressurization system), SIET, SPES Facility in Italy (full-pressure integral passive safety system tests), and Oregon State University (low-pressure integral passive safety system tests).

EPRI has, with a broad participation of numerous countries, developed a Utility Requirements Document (URD) for ALWRs, taking into account the wealth of information related to nuclear power plant safety and operations that has been generated worldwide with commercial nuclear power. The purpose of the URD is to delineate utility desires for their next generation of nuclear plants, and to this end, it consists of a comprehensive set of design requirements for future plants.

Incorporation of the ALWR URD has given the AP-600 a well-defined design basis that is confirmed through thorough engineering analyses and testing and is in conformance with the URD. Some of the high-level design characteristics of the plant are:

- Net electrical power of at least 600 MW(e); and a thermal power of 1940 MWt;
- Rated performance is achieved with up to 10% of the steam generator tubes plugged and with a maximum hot leg temperature of 600°F (315.6°C);
- Core design is robust with at least a 15% operating margin on core power parameters;
- Short lead time (five years from owner's commitment to commercial operation) and construction schedule (3 years);
- No plant prototype is needed since proven power generating system components are used;
- Major safety systems are passive; they require no operator action for 72 hours after an accident, and maintain core and containment cooling for a protracted time without ac power;
- Predicted core damage frequency of 1.7E-07/yr is well below the 1E-05/yr requirement, and frequency of significant release of 3.0E-08/yr is well below the 1E-06/yr requirement;
- Standard design is applicable to anticipated US sites;
- Occupational radiation exposure expected to be below 0.7 man-Sv/yr (70 man-rem/yr);

- Core is designed for a 24-month fuel cycle assuming an 87% capacity factor; capable of an 18-month cycle;
- Refuelling outages can be conducted in 17 days or less;
- Plant design life of 60 years without replacement of the reactor vessel.
- Overall plant availability greater than 93%, including forced and planned outages; the goal for unplanned reactor trips is less than one per year.

5.2.2 Description of the nuclear systems

5.2.2.1 Primary circuit and its main characteristics

The primary circuit of the AP-600 reactor has retained most of the general design features of current designs, but some evolutionary features that enhance the safety and maintainability of the system have been adopted. The coolant loops (Figure 5.2-1) consist of two hot leg and four cold leg pipes, and the reactor coolant pumps are installed directly onto the steam generators, eliminating the primary piping between pumps and steam generator; these features significantly contribute to safety and maintainability. Also, a simplified support structure for the primary systems reduces in-service inspections and improves accessibility for maintenance.

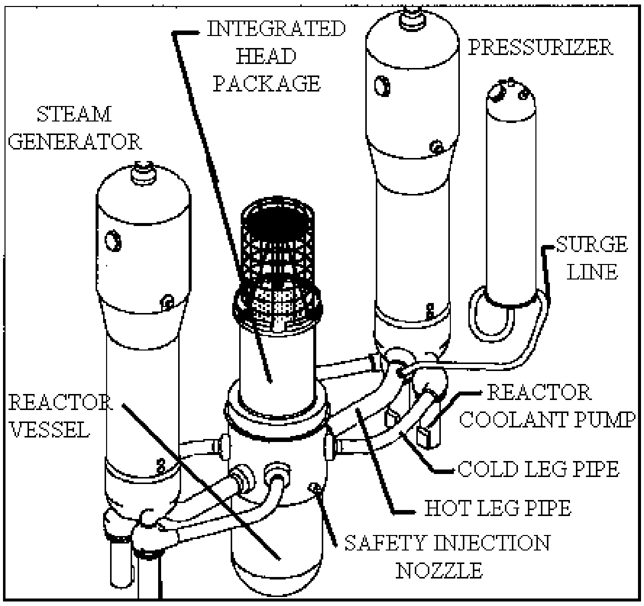


FIG. 5.2-1. Isometric view of NSSS

The reactor coolant system pressure boundary provides a barrier against the release of radioactivity generated within the reactor and is designed to provide a high degree of integrity throughout operation of the plant.

5.2.2.2 Reactor core and fuel design

The core, reactor vessel, and reactor internals of the AP-600 are similar to those of earlier Westinghouse PWRs. Several important features based on existing technology measurably enhance performance characteristics as compared with earlier Westinghouse PWRs. The reactor core is a low-power density design that uses a 12 foot (366 cm), 17×17 fuel assembly. Low-power density is achieved by making the core larger than earlier Westinghouse 600 MW(e) designs, with the number of fuel assemblies increased from 121 to 145. This configuration results in core power density and average linear power density reductions of about 25 percent over earlier Westinghouse plants of the by making the core larger than earlier Westinghouse 600 MW(e) designs, with the number of fuel assemblies increased from 121 to 145. This configuration results in core power density and average linear power density reductions of about 25 percent over earlier Westinghouse plants of the same power rating. This results in lower fuel enrichments, less reliance on burnable absorbers, and longer fuel cycles.

Another feature that contributes to lowering fuel cycle cost and extending reactor life is the use of a stainless steel radial neutron reflector. This reflector reduces neutron leakage, thereby improving core neutron utilization and allowing for reduced fuel enrichment. The radial reflector, see Reactor Internals, has the added benefit of reducing the damaging neutron fluence on the reactor vessel, an important factor in achieving the 60-year design life objective.

The combination of the radial reflector and the low-power density core results in a fuel-cycle cost savings of 15 to 20 percent compared with a standard design of the same power rating.

Another core design feature is the use of reduced-worth control rods (termed "gray" rods) to achieve daily load follow without requiring daily changes in the soluble boron concentration. The use of gray rods, in conjunction with an automated load follow control strategy, eliminates the need for processing thousands of gallons of water per day to change the soluble boron concentration sufficiently to achieve a daily load follow schedule. As a result, systems are simplified through the elimination of the evaporator, and other boron processing equipment (such as pumps, valves, and piping). With the exception of the neutron absorber materials used, the design of the gray rod assembly is identical to that of a normal control rod assembly.

The core consists of three radial regions that have different enrichments; the enrichment of the fuel ranges from 2 to 4%. The temperature coefficient of reactivity of the core is highly negative. The core is designed for a fuel cycle of 18 to 24 months with discharge burnups as high as 55 000 MWd/t.

5.2.2.3 Fuel handling and transfer systems

Refuelling of the reactor is performed in the same way as for current plants. After removing the vessel head, fuel handling takes place from above, using the refuelling machine. During refuelling, one third of the core inventory is replaced.

New fuel storage - New fuel is stored in a high density rack that includes integral neutron absorbing material to maintain the required degree of sub-criticality. The rack is designed to store fuel of the maximum design basis enrichment. The rack in the new fuel pit consists of an array of cells interconnected to each other at several elevations and to supporting grid structures at the top and bottom elevations. The new fuel rack includes storage locations for 56 fuel assemblies. Minimum separation between adjacent fuel assemblies is sufficient to maintain a subcritical array even in the event the building is flooded with un-borated water or fire extinguishing aerosols or during any design basis event.

Spent fuel storage - Spent fuel is stored in high density racks which include integral neutron absorbing material to maintain the required degree of subcriticality. The racks are designed to store fuel of the maximum design basis enrichment. The spent fuel storage racks include storage locations for 619 fuel assemblies. The modified 10×7 rack module additionally contains integral storage locations for five defective fuel storage containers. The design of the rack is such that a fuel assembly can not be inserted into a location other than a location designed to receive an assembly.

5.2.2.4 Primary components

Reactor pressure vessel -The reactor vessel (Figure 5.2-2) is the high pressure containment boundary used to support and enclose the reactor core. The vessel is cylindrical, with a hemispherical bottom head and removable flanged hemispherical upper head

The reactor vessel is approximately 38 feet (11.7 m) long and has an inner diameter at the core region of 157 inches (3.988 m). The total weight of the vessel is approximately 400 tons. Surfaces which can become wetted during operation and refuelling are clad with stainless steel welded overlay. The AP-600 reactor vessel is designed to withstand the design environment of 2500 psia (17.2 MPa) and 650°F (343 ^{o}C) for 60 years. The major factor affecting vessel life is radiation degradation of the lower shell.

As a safety enhancement, there are no penetrations below the top of the core. This eliminates the possibility of a loss of coolant accident by leakage from the reactor vessel which could allow the core to be uncovered. The core is positioned as low as possible in the vessel to limit reflood time in accident situations.

Reactor internals - The reactor internals, the core support structures, the core shroud, the downcomer and flow guiding structure arrangement, and the above-core equipment and structures, are very similar to those in current plants.

The reactor internals consist of two major assemblies - the lower internals and the upper internals. The reactor internals provide the protection, alignment and support for the core, control rods, and gray rods to provide safe and reliable reactor operation.

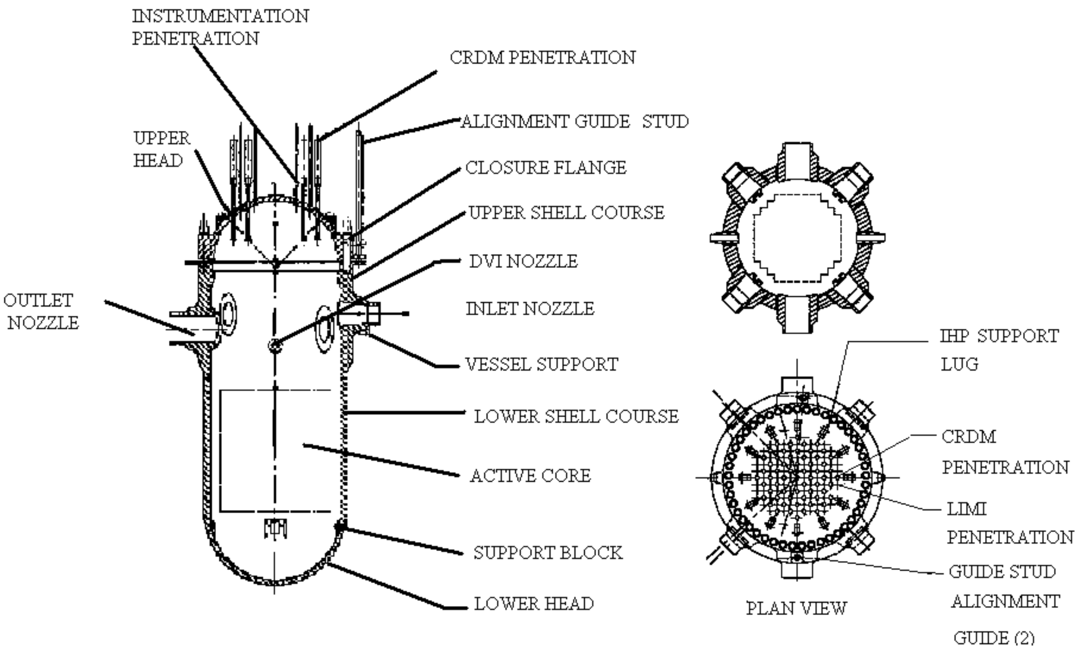


FIG. 5.2-2. Reactor pressure vessel.

The major containment and support member of the reactor internals is the lower core support assembly. This assembly (Figure 5.2-3) consists of the core barrel, lower core support plate, secondary core support, vortex suppression plate, radial reflectors, radial supports, and related attachment hardware. The major material for this structure is 300 series austenitic stainless steel.

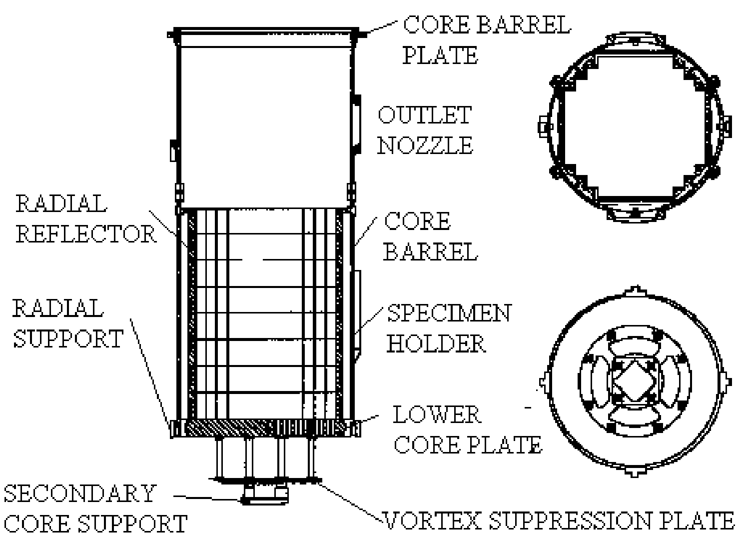


FIG. 5.2-3. Lower core support assembly.

A key feature of the AP-600 plant is the radial reflector. The radial reflector is located inside the core barrel and above the lower core support to form the radial periphery of the core. The reflector provides a transition from the round core barrel to the square fuel assemblies. The reflector assemblies have longitudinal holes that allow cooling water to flow through, while still providing sufficient material to perform the neutron reflection and radiation shielding functions. This results in lower neutron loss from the core and decreased fluence on the reactor pressure vessel. Each reflector assembly (ring) is sized in height so that adjoining sections meet at a fuel grid elevation.

Steam generators - The model Delta-75 steam generator (Figure 5.2-4) is a vertical shell and U-tube evaporator with integral moisture separating equipment based on standard Westinghouse Model F technology. There are currently 84 Model F steam generators operating in 25 nuclear plants with a wide range of operating environments. To date, they have accumulated over 1200 steam-generator-years of operation. The 25 Model F-type replacement steam generators have an impressive record with less than one tube plugged per steam generator for every four years of operation. The basic Delta-75 steam generator design and features have been proven in tests and in previous steam generators including replacement steam generator designs. Design enhancements include nickel-chromium-iron Alloy 690 thermally treated tubes on a triangular pitch, broached tube support plates, improved antivibration bars, single-tier separators, enhanced maintenance features, and a primary-side channel head design that allows for easy access and maintenance by robotic tooling.

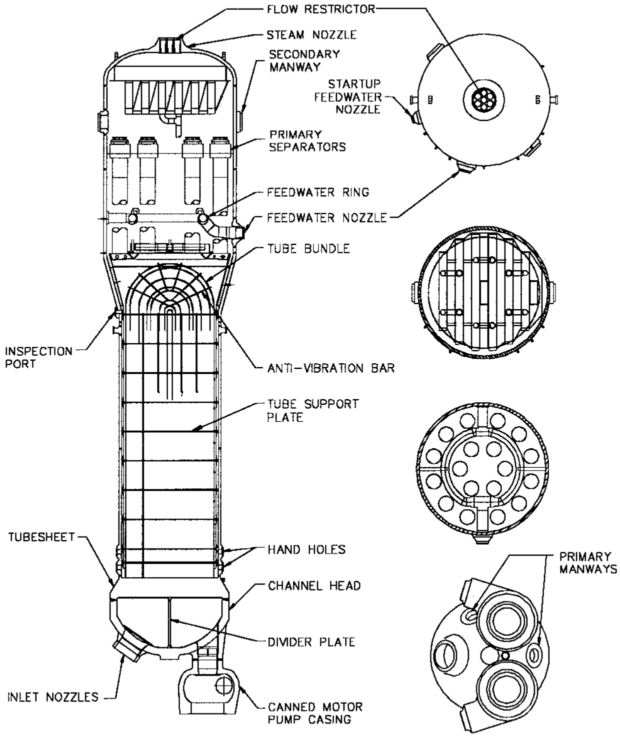


FIG. 5.2-4. Steam generator.

All tubes in the steam generator are accessible for sleeving, if necessary. The design enhancements are based on proven technology. The steam generators operate on all volatile treatment secondary side water chemistry.

Pressurizer - The pressurizer is based on proven Westinghouse technology and years of operating experience. The pressurizer is about 30 percent larger than that normally used in a Westinghouse plant of comparable power rating. The larger pressurizer increases transient operation margins, resulting in a more reliable plant with fewer reactor trips, and avoiding challenges to the plant and operator during transients. It also eliminates the need for fast-acting power-operated relief valves, which are a possible source of RCS leakage and maintenance.

Reactor coolant pumps - The reactor coolant pumps (Figure 5.2-5) are high-inertia, high-reliability, low-maintenance, hermetically sealed canned motor pumps that circulate the reactor coolant through the reactor vessel, loop piping, and steam generators.

The pumps are integrated into the steam generator channel head. The integration of the pump suction into the bottom of the steam generator channel head eliminates the cross-over leg of coolant loop piping; reduces the loop pressure drop; simplifies the foundation and support system for the steam generator, pumps, and piping; and reduces the potential for uncovering of the core by eliminating the need to clear the loop seal during a small loss-of-coolant accident (LOCA). The AP-600 design uses four pumps; two pumps are coupled with each steam generator.

Since the pumps have no seals, they cannot cause a seal failure LOCA. This is a significant safety enhancement, as seal failure LOCA is a major industry issue. Maintenance is also enhanced, since seal replacement is eliminated.

The pumps are mounted in the inverted (motor-below-casing) position. Inverted canned motors have been in operation for over 34 years in marine and fossil boiler circulation systems. These pumps have better operating reliability than upright units because the motor cavity is self-venting into the pump casing, avoiding the potential for gas pockets in the bearing and water regions.

One modification of the pumps from commercial and marine canned motor pump practice is the use of a flywheel to increase the pump rotating inertia. The increased inertia provides a slower rate-of-flow coastdown to improve core thermal margins following the loss of electric power. Extensive testing has validated the manufacturability and operability of the pump flywheel assembly.

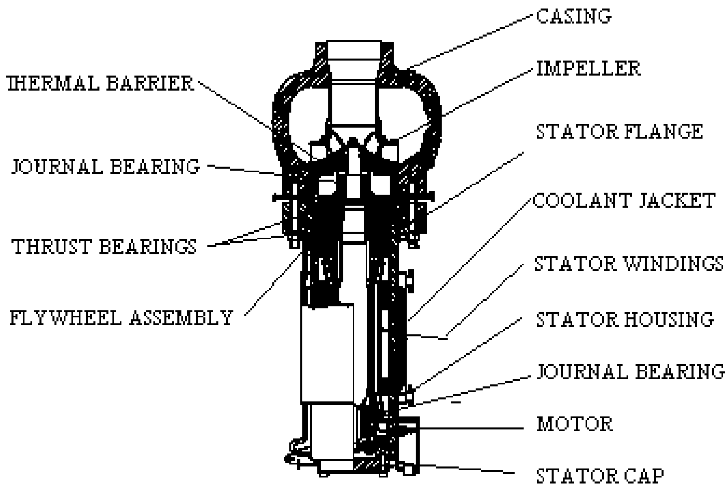


FIG. 5.2-5. *Reactor coolant pump.*

Main coolant lines - Reactor coolant system (RCS) piping is configured with two identical main coolant loops, each of which employs a single 31-inch (790 mm) inside 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 22-inch (560 mm) inside diameter cold leg pipes in each loop (one per pump) transport reactor coolant back to the reactor vessel to complete the circuit.

The RCS loop layout contains several important features that provide for a significantly simplified and safer design. The reactor coolant pumps mount directly on the channel head of each steam generator. This allows the pumps and steam generator to use the same structural support, greatly simplifying the support system and providing more space for pump and steam generator maintenance. The combined steam generator/pump vertical support is a single pinned column extending from the cell floor to the bottom of the channel head. The steam generator channel head is a one-piece forging with manufacturing and inspection advantages over multipiece, welded components. The integration of the pump suction into the bottom of the steam generator channel head eliminates the crossover leg of coolant loop piping, thus avoiding the potential for core uncovery due to loop seal venting after a small loss-of-coolant accident.

The simplified, compact arrangement of the RCS also provides other benefits. The two cold leg lines of the two main coolant loops are identical (except for instrumentation and small line connections) and include bends to provide a low-resistance flow path and flexibility to accommodate the expansion difference between the hot and cold leg pipes. The piping is forged and then bent by a hot induction forming process. The use of a pipe bend reduces costs and in-service inspection requirements by eliminating welds. The loop configuration and material selection yield sufficiently low pipe stresses so that the primary loop and large auxiliary lines meet the requirements to demonstrate leak-before-break. Thus, pipe rupture restraints are not required, greatly simplifying the design and providing enhanced access for maintenance. The simplified RCS loop configuration also allows for a significant reduction in the number of snubbers, whip restraints, and supports. Field service experience and utility feedback have indicated the high desirability of these features.

5.2.2.5 *Reactor auxiliary systems*

Chemical and volume control system - The chemical and volume control system consists of regenerative and letdown heat exchangers, demineralizers and filters, makeup pumps, tanks, and associated valves, piping, and instrumentation.

The chemical and volume control system is designed to perform the following major tasks:

- **Purification** - maintain reactor coolant purity and activity level within acceptable limits;
- **Reactor coolant system inventory control and makeup** - maintain the required coolant inventory in the reactor coolant system; maintain the programmed pressurizer water level during normal plant operations;
- **Chemical shim and chemical control** - maintain the reactor coolant chemistry conditions by controlling the concentration of boron in the coolant for plant startups, normal dilution to compensate for fuel depletion and shutdown boration and provide the means for controlling the reactor coolant system pH by maintaining the proper level of lithium hydroxide;
- **Oxygen control** - provide the means for maintaining the proper level of dissolved hydrogen in the reactor coolant during power operation and for achieving the proper oxygen level prior to startup after each shutdown;
- **Filling and pressure testing of the reactor coolant system** - provide the means for filling and pressure testing of the reactor coolant system. The chemical and volume control system does not perform hydrostatic testing of the reactor coolant system, which is only required prior to initial startup and after major, nonroutine maintenance, but provides connections for a temporary hydrostatic test pump;
- **Borated makeup to auxiliary equipment** - provide makeup water to the primary side systems which require borated reactor grade water;
- **Pressurizer Auxiliary Spray** - provide pressurizer auxiliary spray water for depressurization.

Normal residual heat removal system - The normal residual heat removal system consists of two mechanical trains of equipment, each comprising one pump, one heat exchanger and associated valves, piping and instrumentation. The two trains of equipment share a common suction line from the reactor coolant system and a common discharge header. The major functions of the system are:

- **Shutdown Heat Removal.** The normal residual heat removal system removes both residual and sensible heat from the core and the reactor coolant system. It reduces the temperature of the reactor coolant system during the second phase of plant cooldown. The first phase of cooldown is accomplished by transferring heat from the reactor coolant system via the steam generators to the main steam system;
- Following cooldown, the normal residual heat removal system removes heat from the core and the reactor coolant system during the plant shutdown, until the plant is started up;
- The normal residual heat removal system reduces the temperature of the reactor coolant system from 350° to 120°F (177 to 48.9 °C) within 96 hours after shutdown. The system maintains the reactor coolant temperature at or below 120°F for the plant shutdown;
- **Shutdown Purification.** The normal residual heat removal system provides reactor coolant system flow to the chemical and volume control system during refueling operations;
- **In-Containment Refuelling Water Storage Tank Cooling.** The normal residual heat removal system provides cooling for the in-containment refuelling water storage tank (IRWST) during operation of the passive residual heat removal heat exchanger or during normal plant operations when required. The system is manually initiated by the operator. The normal residual heat removal system limits the IRWST water temperature to less than 212°F (100 °C) during extended operation of the passive residual heat removal system and to not greater than 120°F during normal operation;
- **Low Pressure Reactor Coolant System Makeup and Cooling**. The normal residual heat removal system provides low pressure makeup from the IRWST to the reactor coolant system and provides additional margin for core cooling. The system is manually initiated by the operator following receipt of an automatic depressurization signal. If the system is available, it provides reactor coolant system

makeup once the pressure in the reactor coolant system falls below the shutoff head of the normal residual heat removal system pumps;

- **Low Temperature Overpressure Protection.** The normal residual heat removal system provides a low temperature overpressure protection function for the reactor coolant system during refuelling, startup, and shutdown operations;
- **Long-Term, Post-Accident Containment Inventory Makeup Flowpath.** The normal residual heat removal system provides a flow path for long term post-accident makeup to the reactor containment inventory, under design assumptions of containment leakage.

5.2.2.6 Operating characteristics

The plant control scheme is based on the "reactor follows plant loads". A grid fluctuation can be compensated for through turbine control valves in case of a frequency drop. A decrease in pressure at the turbine would require an increase in reactor power.

The AP-600 is designed to withstand the following operational occurrences without the generation of a reactor trip or actuation of the safety related passive engineered safety systems:

- ± 5%/minute ramp load change within 15% and 100% power,
- ± 10% step load change within 15% and 100% power,
- 100% generator load rejection,
- 100-50-100% power level daily load follow over 90% of the fuel cycle life,
- Grid frequency changes equivalent to 10% peak-to-peak power changes at 2%/minute rate,
- 20% power step increase or decrease within 10 minutes,
- Loss of a single feedwater pump.

The logic and setpoints for all of the AP-600 Nuclear Steam Supply System (NSSS) control systems are developed in order to meet the above operational transients without reaching any of the protection system setpoints.

5.2.3 Description of turbine generator plant system

5.2.3.1 Turbine generator plant

The AP-600 turbine consists of a double-flow, high-pressure cylinder and two double-flow, low-pressure cylinders that exhaust to individual condensers. It is a four flow tandem-compound, 1800 rpm machine. The turbine generator is intended for base load operation and also has load follow capability. Mechanical design of the turbine root and rotor steeple attachments uses optimized contour to significantly reduce operational stresses. Steam flow to the high-pressure turbine is controlled by two floor-mounted steam chests. Each contains two throttle/stop valve assemblies, and two load-governing valves. The single moisture separator reheater (MSR) has a single stage of reheat.

The condenser and circulating water systems have been optimized. The condenser is a twin-shell, multipressure unit with one double-flow, low-pressure turbine exhausting into the top of each shell.

The turbine-generator and associated piping, valves, and controls are located completely within the turbine building and are accessible under operating conditions. There are no safety-related systems or components located within the turbine building. The probability of destructive overspeed condition and missile generation, assuming the recommended inspection frequency, is less than 1×10^{-5} per year. In addition, orientation of the turbine-generator is such that high-energy missiles would be directed away at right angles to safety-related structures, systems, or components. Failure of turbine-generator equipment does not preclude safe shutdown of the reactor. The single direct-driven generator is gas-cooled and rated at 856 MVA at 22 kV, and a power factor of 0.9. Other related system components include a complete turbine-generator bearing lubrication oil system, a digital electrohydraulic (DEH) control system with supervisory instrumentation, a turbine steam sealing system, overspeed protective devices, turning gear, a generator hydrogen and seal oil system, a generator CO_2 system, an exciter cooler, a rectifier section, an exciter, and a voltage regulator.

5.2.3.2 Condensate and feedwater systems

The condensate and feedwater system supplies the steam generators with heated feedwater in a closed steam cycle using regenerative feedwater heating. The condensate and feedwater system is composed of the condensate system, the main and startup feedwater system, and portions of the steam generator system.

The feedwater cycle consists of seven stages of feedwater heating with two parallel string, low-pressure feedwater heaters located in the condenser neck with the next two single-string, low-pressure heaters, deaerator, and the high-pressure heaters located within the turbine building. The condenser hotwell and deaerator storage capacity allows margin in the design. This margin, coupled with three 50 percent condensate pumps, provides greater flexibility and the ability for an operator to control feedwater and condensate transients.

5.2.3.3 Auxiliary systems

Radioactive waste management - The radioactive waste management systems include systems to deal with liquid, gaseous and solid waste. These systems are closely integrated with the chemical and volume control system (CVS).

The liquid waste management systems include the systems that may be used to process for disposal liquids containing radioactive material. These include the following:

- Steam generator blowdown processing system
- Radioactive waste drain system
- Liquid radwaste system

The liquid radwaste system (WLS) uses ion exchangers to process and discharge all wastes from the reactor coolant system. To enhance ion exchange performances, the WLS is divided in two trains to separate borated reactor water from mixed liquid waste.

The gaseous radwaste system is a once-through, ambient-temperature, charcoal delay system. The system consists of a drain pot, a gas cooler, a moisture separator, an activated charcoal-filled guard bed, and two activated charcoal-filled delay beds. Also included in the system are an oxygen analyzer subsystem and a gas sampling subsystem.

The radioactive fission gases entering the system are carried by hydrogen and nitrogen gas. The primary influent source is the liquid radwaste system degasifier. The degasifier extracts both hydrogen and fission gases from the chemical and volume control system letdown flow.

The solid waste management system is designed to collect and accumulate spent ion exchange resins and deep bed filtration media, spent filter cartridges, dry active wastes, and mixed wastes generated as a result of normal plant operation, including anticipated operational occurrences. The system is located in the auxiliary and radwaste buildings. Processing and packaging of wastes are by mobile systems in the auxiliary building loading bay and the mobile systems facility which is a part of the radwaste building. The packaged waste is stored in the annex, auxiliary and radwaste buildings until it is shipped offsite to a licensed disposal facility.

5.2.4 Instrumentation and control systems

The I&C system design for AP-600 integrates individual systems using similar technology. The heart of the system is the portions used for plant protection and for operation of the plant.

The integrated AP-600 I&C system provides the following benefits:

- Control wiring is reduced by 80 percent compared to equivalent hard wired plants without passive safety features;
- Cable spreading rooms are eliminated;
- Duplicate sensors, signal conditioners, and cables are eliminated;
- Maintenance is simplified;
- Plant design changes have little impact on I&C design;
- Accurate, drift-free calibration is maintained;
- Operating margins are improved.

The AP-600 man-machine interfaces have been simplified compared to existing Westinghouse plants. The probability of operator error is reduced and operations, testing, and maintenance are simplified. An automatic signal selector in the control system selects from a redundant sensor for control inputs in lieu of requiring manual selection by the control board operator. Accident monitoring and safety parameters are displayed on safety qualified displays with a co-ordinated set of graphics generated by the qualified data processor. The major benefits of the improved man-machine interfaces are:

- Reduced quantity of manual actions is required,
- Reduced quantity of data is presented to operator,
- Number of alarms is reduced,
- Improved quality of data is presented to operator,
- Data is interpreted for the operator by system computer,
- Maintenance is simplified.

5.2.4.1 Design concept, including control room

The instrumentation and control architecture for the AP-600 is arranged in a hierarchical manner to provide a simplified, structured design that is horizontally and vertically integrated.

Above the monitor bus are the systems that facilitate the interaction between the plant operators and the I&C. These are the operations and control centers system (OCS) and the data display and monitoring system (DDS). Below the monitor bus are the systems and functions that perform the protective, control, and data monitoring functions.

These are the protection and safety monitoring system (PMS) (Section 5.2.4.2) the plant control system (PLS), the special monitoring system (SMS), the diverse actuation system and the in-core instrumentation system (IIS).

The plant control system (PLS) has the function of establishing and maintaining the plant operating conditions within prescribed limits. The control system improves plant safety by minimizing the number of situations for which some protective response is initiated and relieves the operator from routine tasks.

The purpose of the diverse actuation system (DAS) is to provide alternative means of initiating the reactor trip and emergency safety features. The hardware and software used to implement the DAS are different from the hardware and software used to implement the protection and safety monitoring system. The DAS is included to meet the anticipated transient without (reactor) trip (ATWT) rule and to reduce the probability of a severe accident resulting from the unlikely coincidence of a transient and common mode failure of the protection and safety monitoring. The protection and safety monitoring system is designed to prevent common mode failures; however, in the low-probability case where a common mode failure could occur, the DAS provides diverse protection.

Main control room - The operations and control centers system includes the complete operational scope of the main control room, the remote shutdown workstation, the waste processing control room, and partial scope for the technical support center. With the exception of the control console structures, the equipment in the control room is part of the other systems (for example, protection and safety monitoring system, plant control system, data and display processing system). The conceptual arrangement of the main control room is shown in Figure 5.2-6.

The boundaries of the operations and control center system for the main control room and the remote shutdown workstation are the signal interfaces with the plant components. These interfaces are via the plant protection and safety monitoring system processor and logic circuits, which interface with the reactor trip and engineered safety features plant components; the plant control system processor and logic circuits, which interface with the non-safety-related plant components; and the plant monitor bus, which provides plant parameters, plant component status, and alarms.

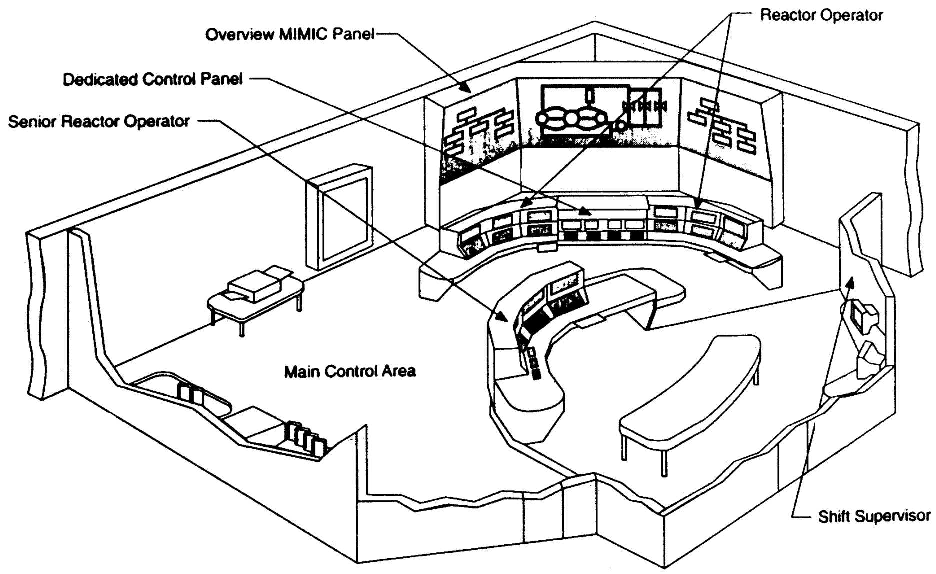


FIG. 5.2-6. Main control room.

5.2.4.2 *Reactor protection system and other safety systems*

The AP-600 provides instrumentation and controls to sense accident situations and initiate engineered safety features. The occurrence of a limiting fault, such as a loss-of-coolant accident or a secondary system break, requires a reactor trip plus actuation of one or more of the engineered safety features. This combination of events prevents or mitigates damage to the core and reactor coolant system components, and provides containment integrity.

The protection and safety monitoring system (PMS) provides the safety-related functions necessary to control the plant during normal operation, to shut down the plant, and to maintain the plant in a safe shutdown condition. The protection and safety monitoring system controls safety-related components in the plant that are operated from the main control room or remote shutdown workstation.

5.2.5 Electrical systems

The AP-600 on-site power system includes the main AC power system and the DC power system. The main AC power is a non-Class 1E system. The DC power system consists of two independent systems, one Class 1E and one non-Class 1E. The on-site power system is designed to provide reliable electric power to the plant safety and non-safety equipment for normal plant operation, startup, and normal shut down, and for accident mitigation and emergency shutdown.

The main generator is connected to the off-site power system via three single-phase main step-up transformers. The normal power source for the plant auxiliary AC loads is provided from the 22 kV isophase generator buses through the two unit auxiliary transformers of identical ratings. In the event of a loss of the main generator, the power is maintained without interruption from the preferred power supply by an auto-trip of the main generator breaker. Power then flows from the transformer area to the auxiliary loads through the main and unit auxiliary transformers.

Off-site power has no safety-related function due to the passive safety features incorporated in the AP-600 design. Therefore, redundant off-site power supplies are not required. The design provides a reliable offsite power system that minimizes challenges to the passive safety system.

5.2.5.1 Operational power supply systems

The main AC power system is a non-Class 1E system that does not perform any safety function. The standby power supply is included in the on-site standby power system.

The power to the main AC power system normally comes from the station main generator through unit auxiliary transformers. The plant is designed to sustain a load rejection from 100 percent power with the turbine generator continuing stable operation while supplying the plant house loads. The load rejection feature does not perform any safety function

The on-site standby AC power system is powered by the two on-site standby diesel generators and supplies power to selected loads in the event of loss of normal, and preferred AC power supplies.

The plant DC power system comprises two independent Class 1E and non-Class 1E DC power systems. Each system consists of ungrounded stationary batteries, DC distribution equipment, and uninterruptible power supplies.

5.2.5.2 Safety-related systems

The Class 1E DC power system includes four independent divisions of battery systems. Any three of the four divisions can shut down the plant safely and maintain it in a safe shutdown condition. Divisions B and C have two battery banks. One of these battery banks is sized to supply power to selected safety-related loads for at least 24 hours, and the other battery bank is sized to supply power to another set of selected safety-related loads for at least 72 hours following a design basis event (including the loss of all AC power).

For supplying power during the post-72 period, provisions are made to connect an ancillary ac generator to Class 1E voltage regulating transformers (Divisions B and C only).

5.2.6 Safety concepts

5.2.6.1 Safety requirements and design philosophy

The AP-600 design provides for multiple levels of defense for accident mitigation (defense-in-depth), resulting in extremely low core damage probabilities while minimizing the occurrences of containment flooding, pressurization, and heat-up situations. This defense-in-depth capability includes multiple levels of defense for a very wide range of plant events. Defense-in-depth is integral to the AP-600 design, with a multitude of individual plant features capable of providing some degree of defense of plant safety. Six aspects of the AP-600 design contribute to defense-in-depth:

Stable Operation. In normal operation, the most fundamental level of defense-in-depth ensures that the plant can be operated stably and reliably. This is achieved by the selection of materials, by quality assurance during design and construction, by well-trained operators, and by an advanced control system and plant design that provide substantial margins for plant operation before approaching safety limits.

Physical Plant Boundaries. One of the most recognizable aspects of defense-in-depth is the protection of public safety through the physical plant boundaries. Releases of radiation are directly prevented by the fuel cladding, the reactor pressure boundary, and the containment pressure boundary.

Passive Safety-Related Systems. The highest level of defense includes the AP-600 safety-related passive systems and equipment. The safety-related passive systems are sufficient to automatically establish and maintain core cooling and containment integrity for the plant following design basis events, assuming that the most limiting single failure occurs. These systems maintain core cooling and containment integrity after an event, without operator action and onsite and offsite ac power sources, for an indefinite amount of time (see section 5.2.6.2 for a description of AP-600 passive safety systems).

Diversity within the Safety-Related Systems. An additional level of defense is provided through the diverse mitigation functions within the passive safety-related systems themselves. This diversity exists, for example, in the residual heat removal function. The PRHR HX is the passive safety-related feature for removing decay heat during a transient. In case of multiple failures in the PRHR HX, defense-in-depth is provided by the passive safety injection and automatic depressurization (passive feed and bleed) functions of the PXS.

Non-safety Systems. The next level of defense-in-depth is the availability of certain non-safety systems for reducing the potential for events leading to core damage. For more probable events, these defense-in-depth non-safety systems automatically actuate to provide a first level of defense to reduce the likelihood of unnecessary actuation and operation of the safety-related systems. These non-safety-related systems establish and maintain safe shutdown conditions for the plant following design basis events, provided that at least one of the nonsafety-related ac power sources is available.

Containing Core Damage. The AP-600 design provides the operators with the ability to drain the in-containment refuelling water storage tank (IRWST) water into the reactor cavity in the event that the core has uncovered and is melting. The objective of this cavity flooding action is to prevent reactor vessel failure and subsequent relocation of molten core debris into the containment. Retention of the debris in the vessel significantly reduces the uncertainty in the assessment of containment failure and radioactive release to the environment due to ex-vessel severe accident phenomena.

AP-600 defense-in-depth features enhance safety such that no severe release of fission products is predicted to occur from an initially intact containment for more than 100 hours after the onset of core damage, assuming no actions for recovery. This amount of time provides for performance of accident management actions to mitigate the accident and prevent containment failure. The frequency of severe release as predicted by PRA is 3.0E-08 per reactor year.

5.2.6.2 Safety systems and features (active, passive and inherent)

The use of passive safety systems provides significant and measurable improvements in plant simplification, safety, reliability, and investment protection. The AP-600 uses passive safety systems to improve the safety of the plant and to satisfy safety criteria of regulatory authorities. 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 machinery 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 safety positions upon loss of power or upon receipt of a safety actuation signal. However, they are also supported by multiple, reliable power sources to avoid unnecessary actuations.

The AP-600 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.

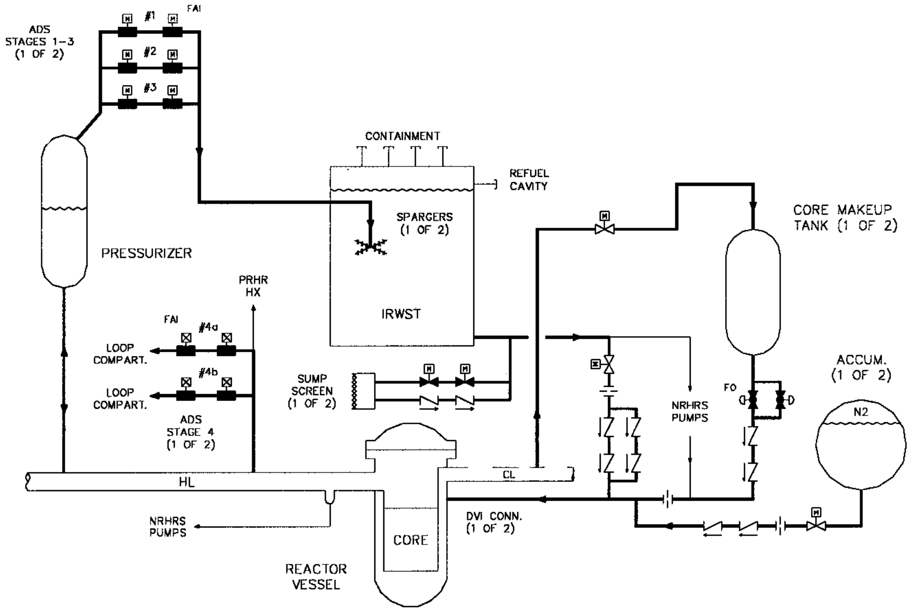


FIG. 5.2-7. Passive core cooling system.

These passive safety systems provide a high degree of plant safety and investment protection. They establish and maintain core cooling and containment integrity indefinitely, with no operator or ac power support requirements. The passive systems are designed to meet the single-failure criteria, and probabilistic risk assessments (PRAs) are used to verify their reliability.

The AP-600 passive safety systems are significantly simpler than typical PWR safety systems since they contain significantly fewer components, reducing the required tests, inspections, and maintenance, require no active support systems, and their readiness is easily monitored. The passive safety systems have one-third the number of remote valves as typical active safety systems, and they contain no pumps. In addition, these passive safety systems do not require a radical departure in the design of the rest of the plant, core, RCS, or containment. This design approach allows the licensing safety criteria to be satisfied with a greatly simplified plant design.

Emergency core cooling system - The passive core cooling system (PXS) **(**Figure 5.2-7) 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 US NRC-approved codes) demonstrate the effectiveness of the PXS in protecting the core following various RCS break events, even for breaks as severe as the 8-inch (200 mm) vessel injection lines. The PXS provides approximately a 400°F (220°C) 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 uses three passive sources of water to maintain core cooling through safety injection. These injection sources include the core makeup tanks (CMTs), the accumulators, and the in-containment refuelling water storage tank (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.

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. As a result, the RCS must be depressurized before injection can occur.

The depressurization of the RCS is automatically controlled to reduce pressure to about 12 psig (0.18 MPa); 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 the four stages of the ADS to permit a relatively slow, controlled RCS pressure reduction.

Passive residual heat removal - The PXS includes a 100% capacity passive residual heat removal heat exchanger (PRHR HX). The PRHR HX is connected through inlet and outlet lines to RCS loop 1. The PRHR HX protects the plant against transients that upset the normal steam generator feedwater and steam systems. The PRHR HX satisfies the safety criteria for loss of feedwater, feedwater line breaks, and steam line breaks.

The IRWST provides the heat sink for the PRHR HX. 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. This steam condenses on the steel containment vessel and, after collection, drains by gravity back into the IRWST. The PRHR HX and the passive containment cooling system provide indefinite decay heat removal capability with no operator action required.

Passive containment cooling system - The passive containment cooling system (PCS) (Figure 5.2-8) 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.

The steel containment vessel 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 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.

Westinghouse has calculated the AP-600 to have a significantly reduced frequency of release of large amounts of radioactivity following a severe accident core damage scenario. This analysis shows that with only the normal PCS air cooling, the containment stays well below the predicted failure pressure. Other factors include improved containment isolation and reduced potential for LOCAs outside of containment. This improved containment performance supports the technical basis for simplification of offsite emergency planning.

Main control room habitability system - The main control room habitability system (VES) provides fresh air, cooling, and pressurization to the main control room (MCR) following a plant accident. Operation of the VES is automatically initiated upon receipt of a high MCR radiation signal, which isolates the normal control room ventilation path and initiates pressurization. Following system actuation, all functions are completely passive.

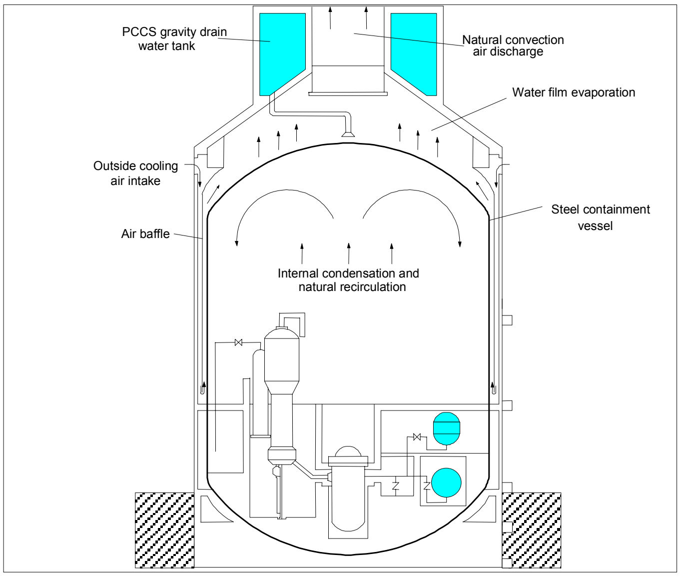


FIG. 5.2-8. Passive containment cooling system.

The VES air supply is contained in a set of compressed air storage tanks. The VES also maintains the MCR at a slight positive pressure, to minimize the infiltration of airborne contaminants from the surrounding areas.

Containment isolation - AP-600 containment isolation is significantly improved. One major improvement is the large reduction in the number of penetrations. Furthermore, the number of normally open penetrations is reduced by 60 percent. There are no penetrations required to support post-accident mitigation functions (the canned motor reactor coolant pumps do not require seal injection, and the passive residual heat removal and passive safety injection features are located entirely inside containment).

Long-term accident mitigation - A major safety advantage of the AP-600 is that long-term accident mitigation is maintained without operator action and without reliance on offsite or onsite ac power sources. The passive safety systems are designed to provide long-term core cooling and decay heat removal without the need for operator actions and without reliance on the active nonsafety-related systems. For the limiting design basis accidents, the core coolant inventory in the containment for recirculation cooling and boration of the core is sufficient to last for at least 30 days, even if inventory is lost at the design basis containment leak rate.

5.2.6.3 Severe accidents (Beyond design basis accidents)

In-vessel retention of molten core debris

In-vessel retention (IVR) of core debris by cooling from the outside is a severe accident mitigation attribute of the AP-600 design. With the reactor vessel intact and debris retained in the lower head, phenomena that may occur as a result of core debris being relocated to the reactor cavity are prevented. The AP-600 is provided with reactor vessel insulation that promotes in-vessel retention and surface treatment that promotes wetability of the external surface.

The design features of the AP-600 containment promote flooding of the containment cavity region during accidents, submerging of the reactor vessel lower head in water. Liquid effluent released through the break during a LOCA event is directed to the reactor cavity. The AP-600 design also includes a provision for draining the in-containment refuelling water storage tank (IRWST) water into the reactor cavity through an operator action.

5.2.7 Plant layout

5.2.7.1 Buildings and structures, including plot plan

A typical site plan for a single unit AP-600 is shown on Figure 5.2-9. The power block complex consists of five 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 are constructed on individual basemats. The nuclear island consists of the containment building, the shield building, and the auxiliary building, all of which are constructed on a common basemat.

The plant arrangement contains conventional and unique features that facilitate and simplify operational and maintenance activities. For example, accessibility to the containment during an outage is extremely important to those maintenance activities that can be performed only during the outage. The AP-600 containment contains a 22-foot (6.7 m) diameter main equipment hatch and a personnel airlock at the operating deck level, and a 16-foot (4.9 m) diameter maintenance hatch and a personnel airlock at grade level. These large hatches significantly enhance accessibility to the containment during outages and, consequently, reduce the potential for congestion at the containment entrances. These containment hatches, located at the two different levels, allow activities occurring above the operating deck to be unaffected by activities occurring below the operating deck.

The containment arrangement provides a large laydown areas inside containment at both the operating deck level and the maintenance floor level. Additionally, the auxiliary building and the adjacent annex building provide large staging and laydown areas immediately outside of both large equipment hatches.

The AP-600 consists of the following five principal structures. Each of these buildings are constructed on individual basemats:

- Nuclear island (Containment, Shield building, Auxiliary building),
- Annex building,
- Diesel generator building,
- Radwaste building,
- Turbine building.

5.2.7.2 Reactor building

The reactor building of the AP-600 basically coincides with the shield building surrounding the containment (Section 5.2.7.3).

5.2.7.3 Containment

Containment building - The containment building is the containment vessel and all structures contained within the containment vessel. The containment building is an integral part of the overall containment system with the functions of containing the release of airborne radioactivity following postulated design basis accidents and providing shielding for the reactor core and the reactor coolant system during normal operations.

The containment vessel 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.

The principal system located within the containment building is the reactor coolant system, the passive core cooling system, and the reactor coolant purification portion of the chemical and volume control system.

Shield building - The shield building is the structure and annulus area that surrounds the containment building. During normal operations, a primary function of the shield building is to provide shielding for the containment vessel and the radioactive systems and components located in the containment building. The shield building, in conjunction with the internal structures of the containment building, provides the required shielding for the reactor coolant system and all the other radioactive systems and components housed in the containment. During accident conditions, the shield building provides the required shielding for radioactive airborne materials that may be dispersed in the containment as well as radioactive particles in the water distributed throughout the containment.

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. It is attached to the cylindrical section of the containment vessel. The function of the passive containment cooling system air baffle is to provide a pathway for natural circulation of cooling air in the event that a design basis accident results in a large release of energy into the containment. In this event the outer surface of the containment vessel transfers heat to the air between the baffle and the containment shell. This heated and thus, lower density air flows up through the air baffle to the air diffuser and cooler and higher density air is drawn into the shield building through the air inlet in the roof of the shield building.

1. CONTAINMENT/SHIELD BUILDING
2. TURBINE BUILDING
3. ANNEX BUILDING
4. AUXILIARY BUILDING
5. RADWASTE BUILDING
6. DIESEL GENERATOR BUILDING
7. PUMP BASIN
8. TURBINE LUBE OIL DRAIN TANK
9. DIESEL GENERATOR FUEL OIL STORAGE TANK
10. DEMINERALIZED WATER STORAGE TANK

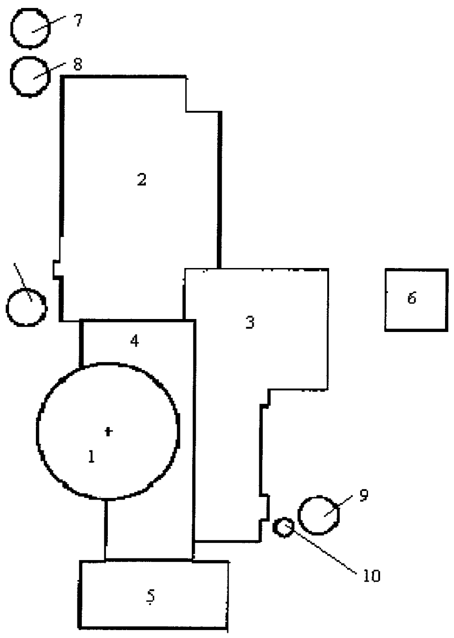


FIG. 5.2-9. AP-600 - Site layout.

Another function of the shield building is to protect the containment building from external events. The shield building protects the containment vessel and the reactor coolant system from the effects of tornadoes and tornado produced missiles.

5.2.7.4 Turbine building

The turbine building houses the main turbine, generator, and associated fluid and electrical systems. It provides weather protection for the laydown and maintenance of major turbine/generator components. The turbine building also houses the makeup water purification system. No safety-related equipment is located in the turbine building.

5.2.7.5 *Other buildings*

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. The auxiliary building provides protection for the safety-related equipment against the consequences of either a postulated internal or external event and also provides shielding for the radioactive equipment and piping that is housed within the building.

The most significant equipment, systems and functions contained within the auxiliary building are the following:

Main control room. The main control room provides the man-machine interfaces required to operate the plant safely under normal conditions and to maintain it in a safe condition under accident conditions.

Instrumentation and control systems. The instrumentation and control systems provide monitoring and control of the plant during startup, ascent to power, powered operation, and shutdown.

Class 1E direct current system. The Class 1E DC system provides 125 volts power for safety-related and vital control instrumentation loads including monitoring and control room emergency lighting.

Fuel handling area. The primary function of the fuel handling area is to provide for the handling and storage of new and spent fuel.

Mechanical equipment areas. The mechanical equipment located in radiological control areas of the auxiliary building are the normal residual heat removal pumps and heat exchangers, the spent fuel cooling system pumps and heat exchangers, the liquid and gaseous radwaste pumps, tanks, demineralizers and filters, the chemical and volume control pumps and heating, ventilating and air conditioning exhaust fans.

Containment penetration areas. The auxiliary building contains all of the mechanical, electrical and I&C containment penetration areas from the shield building.

Main steam and feedwater isolation valve compartment. The main steam and feedwater isolation valve compartment is contained within the auxiliary building. The auxiliary building provides an adequate venting area for the main steam and feedwater isolation valve compartment in the event of a postulated leak in either a main steam line or feedwater line.

Annex building - The annex building provides the main personnel entrance to the power generation complex. It includes access ways for personnel and equipment to the clean areas of the nuclear island in the auxiliary building and to the radiological control area. The building includes the health physics facilities for the control of entry to and exit from the radiological control area as well as personnel support facilities such as locker rooms.

Diesel generator building - The diesel generator building houses two identical slide along diesel generators separated by a three hour fire wall. These generators provide backup power for plant operation in the event of disruption of normal power sources. No safety-related equipment is located in the diesel generator building.

Radwaste building - The radwaste building includes facilities for segregated storage of various categories of waste prior to processing, for processing by mobile systems, and for storing processed waste in shipping and disposal containers. No safety-related equipment is located in the radwaste building.

5.2.8 Technical data

General plant data

Power plant output, gross	619	MW(e)
Power plant output, net	600	MW(e)
Reactor thermal output [core power 1933 MWt]	1 940	MWt
Power plant efficiency, net	31	%
Cooling water temperature	30.5	^oC

Nuclear steam supply system

Number of coolant loops	2 hot legs/4 cold legs	
Primary circuit volume, including pressurizer	239	m^3
Steam flow rate at nominal conditions	1 063	kg/s
Feedwater flow rate at nominal conditions	1 063	kg/s
Steam temperature/pressure	272.7/5.74	^oC/MPa
Feedwater temperature/pressure	224/7.21	^oC/MPa

Reactor coolant system

Primary coolant flow rate	9 940	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	279.5	^oC
Coolant outlet temperature, at RPV outlet	315.6	^oC
Mean temperature rise across core	36.1	^oC

Reactor core

Active core height	3.658	m
Equivalent core diameter	2.921	m
Heat transfer surface in the core	4 170	m^2
Fuel inventory		66.9 t U
Average linear heat rate	13.5	kW/m
Average fuel power density	28.89	kW/kg U
Average core power density (volumetric)	78.82	kW/l
Thermal heat flux, F_q	2.60	
Enthalpy rise, F_H	1.65	

Fuel material	Sintered UO_2	
Fuel assembly total length	4 326	mm
Rod array	square, 17×17	
Number of fuel assemblies	145	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	25	
Number of structural spacer grids	9	
Number of intermediate flow mixing grids	4	
Enrichment (range) of first core	1.9-3.7	Wt% U-235
Enrichment of reload fuel at equilibrium core	4.8	Wt% U-235
Operating cycle length (fuel cycle length)	24	months
Average discharge burnup of fuel (nominal)	55 000	MWd/t
Cladding tube material	Zircaloy	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly	664.5	kg
Active length of fuel rods	4 094	mm
Burnable absorber, strategy/material	Wet annular burnable absorber, Integral fuel burnable absorber	
Number of control rods	61 (45 black, 16 gray)	
Absorber rods per control assembly	24	
Absorber material	Ag-In-Cd (black), Ag-In-Cd/304SS (gray)	
Drive mechanism	Magnetic jack	
Positioning rate [in steps/min or mm/s]	45	steps/min
Soluble neutron absorber	Boric acid	

Reactor pressure vessel

Cylindrical shell inner diameter	3 988	mm
Wall thickness of cylindrical shell	203	mm
Total height	11 708	mm
Base material: cylindrical shell	Carbon steel	

RPV head	Carbon steel	
liner	Stainless steel	
Design pressure/temperature	17.2/ 343	MPa/°C
Transport weight (lower part), and	283.3	t
RPV head	79.5	t

<u>Steam generators</u>

Type	Delta 75, vertical, U-tube	
Number	[Thermal capacity 970 MWt] 2	
Heat transfer surface	6 986	m^2
Number of heat exchanger tubes	6 307	
Tube dimensions	17.5/15.5	mm
Maximum outer diameter	4500.8	mm
Total height	21051	mm
Transport weight	365.5	t
Shell and tube sheet material	Carbon steel	
Tube material	Inconel 690-TT	

<u>Reactor coolant pump</u>

Type	Canned motor	
Number		4
Design pressure/temperature	17.2/343.3	MPa/°C
Design flow rate (at operating conditions)	2 485	kg/s
Pump head	73	m
Power demand at coupling, cold/hot	2 240	kW
Pump casing material		
Pump speed		rpm

<u>Pressuriser</u>

Total volume	45.31	m^3
Steam volume: full power/zero power	14.16	m^3
Design pressure/temperature	17.2/360	MPa/°C
Heating power of the heater rods	1600	kW
Number of heater rods		
Inner diameter	354	mm
Total height		mm
Material		
Transport weight		t

<u>Pressuriser relief tank</u> Not applicable

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

<u>Primary containment</u>

Type	Dry, free standing, steel	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	39.6/57.6	m
Free volume		m^3
Design pressure/temperature (DBEs)	0.316/137.8	kPa/°C
(severe accident situations)	0.316 /137.8	kPa/°C
Design leakage rate	0.12	vol%/day
Is secondary containment provided?	No	

<u>Reactor auxiliary systems</u>

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

<u>Power supply systems</u>

Main transformer,	rated voltage	22/	kV
	rated capacity	870	MVA
Plant transformers,	rated voltage	22/4.16	kV
	rated capacity	45	MVA
Start-up transformer	rated voltage	-/4.16	kV
	rated capacity	45	MVA

Medium voltage busbars (6 kV or 10 kV)	6	
Number of low voltage busbar systems	10	
Standby diesel generating units: number	2	
rated power	4	MW
Number of diesel-backed busbar systems	2	
Voltage level of these	4160	V ac
Number of DC distributions	10	
Voltage level of these	125	V dc
Number of battery-backed busbar systems	11	
Voltage level of these	125	V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	Tandem-compound, 4-flow, 47 in. (1200 mm) last-stage blade	
Number of turbine sections per unit	1HP/ 2LP	
Turbine speed	1 800	rpm
Overall length of turbine unit	30	m
Overall width of turbine unit	9	m
HP inlet pressure/temperature	5.6/271.4	MPa/°C

Generator

Type	3-phase, synchronous	
Rated power	880	MVA
Active power	675	MW
Voltage	22	kV
Frequency	60	Hz
Total generator mass [1,216,000 lbs]	552	t
Overall length of generator	18	m

Condenser

Type	Multipressure	
Number of tubes	50 600	
Heat transfer area	73 784	m^2
Cooling water flow rate	24.36	m^3/s
Cooling water temperature	30.5	°C
Condenser pressure	9.1	kPa

Condensate pumps

Number	3	
Flow rate	389	kg/s
Pump head	274	m
Temperature	46	°C
Pump speed	1190	rpm

Condensate clean-up system

Full flow/part flow	part flow, 33%	
Filter type	Deep bed	

Feedwater tank

Volume	284	m^3
Pressure/temperature	1.11/184	MPa/ °C

Feedwater pumps

Number	2	
Flow rate	590	kg/s
Pump head	783	
Feedwater temperature	184	°C
Pump speed	4300	rpm

Condensate and feedwater heaters

Number of heating stages	7	
Redundancies	Two strings for lowest two stage	

5.2.9 Measures to improve economy and maintainability

AP-600 economics is derived from simplification and state-of-the-art construction techniques.

Simplification - AP-600 is an advance passive nuclear power plant that has been designed to meet globally recognized requirements. A concerted effort has been made to simplify systems and components to facilitate construction, operation and maintenance and to reduce the capital and generating costs. The use of passive systems allows the plant design to be significantly simpler than current generation pressurized water reactor plants. In addition, the passive safety systems do not require the large network of safety support systems found in current generation nuclear power plants (e.g., Class 1E ac power, safety HVAC, safety cooling water systems and associated seismic buildings). The AP-600 uses 50% fewer valves, 80% less pipe (safety grade), 70% less control cable, 35% fewer pumps, and 45% less seismic building volume than an equivalent conventional reactor. Simplicity reduces the cost for reasons other than reduction of the number of items to be purchased. With a fewer number of components, the overall cost of installation is reduced and construction time is minimized.

Construction Schedule - The AP-600 has been designed to make use of modern modular construction techniques. Not only does the design incorporate vendor designed skids and equipment packages, it also includes large structural modules and special equipment modules. Modularization allows construction tasks that were traditionally performed in sequence to be completed in parallel. The modules, constructed in factories, can be assembled at the site for a planned construction schedule of 3 years – from ground-breaking to fuel load. This duration has been verified by experienced construction managers through 4D (3D models plus time) reviews of the construction sequence.

Availability and O&M Costs - The AP-600 combines proven Westinghouse PWR technology with utility operating experience to enhance reliability and operability. Model F steam generators, canned motor pumps and rugged turbine generators are proven performers with outstanding operating records. The digital on-line diagnostic instrumentation and control system features an integrated control system, which avoids reactor trips due to single channel failure. In addition, the plant design provides large margins for plant operation before reaching the safety limits. This assures a stable and reliable plant operation with a reduced number of reactor trips (less than one reactor trip per year). Based on the above, and considering the short planned refueling outage (17 days) and plans to use a 24-month fuel cycle, the AP-600 is expected to largely exceed the 93% availability goal. For AP-600 availability is enhanced by the simplicity designed into the plant, as described above. There are fewer components which result in lower maintenance costs, both planned and unplanned. In addition, the great reduction in safety-related components results in a large reduction in inspection and tests. Simplicity is also reflected in the reduced AP-600 staffing requirements.

5.2.10 Project status and planned schedule

The AP-600 design received Final Design Approval from the NRC in 1998 and Design Certification in 1999. The AP-600 is available for construction as a standard plant on commercial basis with no further development needed. The licensing procedure in the US has been greatly simplified. A combined construction permit and operating license for a standard design plant has been agreed by the NRC. The next and final milestone (in the US) is the application for (and issue of) a site specific, combined license by a power company wishing to construct and operate the plant on either a pre-licensed site or a new one.

5.3 HSBWR (HITACHI LTD, JAPAN)

5.3.1 Introduction

The HSBWR (Hitachi Simplified BWR) being developed by Hitachi is a design concept for a Boiling Water Reactor (BWR) in the small and medium size range. This concept aims at establishing a series of BWRs up to a capacity of 600 MW(e). The HSBWR adopts natural circulation of primary coolant and passive safety systems to improve the economy, maintainability, and reliability by simplification. The HSBWR is characterized by:

1. System simplification by adopting natural circulation for coolant recirculation,
2. A reactor building standardization with high seismic resistance,
3. High safety standards with an automatic depressurization system (ADS) and accumulators at low pressure for emergency core cooling systems (ECCSs) and an outer pool for decay heat removal, and
4. A short construction period by adoption of a steel structure primary containment vessel.

The main features of the HSBWR with respect to these characteristics are delineated below:

1. The fuel assemblies are short compared to those in existing BWR plants, with a heated length of 3.1 m and a total length of 3.7 m, to avoid seismic resonance between the core and the reactor building. The latter may be constructed on soft to firm ground, which even makes it possible to construct the standardized plant underground with a high level of protection, if necessary;
2. The volumetric power density of the core is low, only 34.2 kW/l, and the operation period between refuellings is 23 months;
3. The reactor internals arrangement is simple, without forced recirculation systems (i.e. the reactor operates with natural circulation of the reactor coolant) and steam separators;
4. There will be no core uncovery in any loss-of-coolant accident (LOCA) situation with coolant left in the reactor pressure vessel (RPV) and coolant injected by the actuation of the steam driven reactor core isolation cooling (RCIC) systems. The automatic depressurization system (ADS) and the accumulators at a relatively low pressure of 0.5 - 1.0 MPa;
5. The decay heat is absorbed in the suppression pool for one day after accident initiation, and heat conduction through the steel-walled primary containment vessel (PCV) to the surrounding outer pool provides heat removal from the containment for three days, giving operators ample time for recovery actions;
6. The flooding of the RPV by coolant fed from the accumulators through the RPV and direct cooling of its outer surface will help to keep the core covered with coolant and maintain core cooling, even in an RPV bottom break accident;
7. Depressurization by the ADS and injection of borated water from the accumulator at a pressure of 0.5 – 1.0 MPa to decrease reactivity and shut down the reactor in an anticipated transient without scram (ATWS) situation;
8. The primary containment vessel, reactor building and turbine building are standardized and compact, and the same plant layout will apply at any reactor site; and
9. A shortened construction period of 32 - 36 months (depending on the site conditions) including pre-operation and start-up tests will help ensure the economic competitivity.

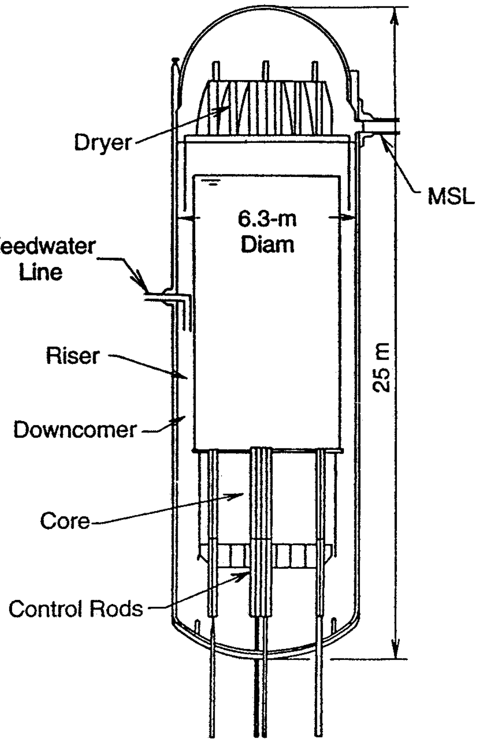


FIG. 5.3-1. The HSBWR reactor vessel arrangement.

5.3.2 Description of the nuclear systems

5.3.2.1 Primary circuit and its main characteristics

The schematic of the reactor pressure vessel (RPV) and the system configuration are shown in Figures 5.3-1 and 5.3-2, respectively.

Natural circulation is used for coolant circulation, i.e., pumped recirculation systems are eliminated, in order to reduce the number of components driven by external force and thereby improve the reliability and maintainability. A riser of 9 m height is installed above the core in order to increase the driving force for natural circulation. The 3.7 m length of the fuel assemblies, with an active length of 3.1 m, is determined by the objective of avoiding seismic resonance between the fuel bundles and the reactor building, independent of the conditions of the ground (soft to firm ground) on which it is constructed. This makes it possible to standardize the reactor building and plant layout without any connection to ground firmness.

Elimination of the steam separators results in a reduced flow resistance in the primary circuit, and the natural circulation flow rate becomes higher. The steam passes from the core and up the riser of 9 m height in 6 seconds, and the passage through the long steam dome takes 8 seconds. The long passing time decreases the strength of gamma-radiation from nitrogen N-16 and reduces the need for gamma shields in the turbine building.

The main coolant lines consist of two main steam lines of 700 mm pipe diameter with two main steam isolation valves (MSIVs), and two feedwater lines of 400 mm pipe diameter with isolation valves

5.3.2.2 Reactor core and fuel design

The power density of the reactor core is lower in a natural circulation reactor than in a forced circulation reactor, but the lower power density allows a longer continuous operation. The short heated length of the fuel and the low power density provide good thermal-hydraulic characteristics. With the 8x8 type fuel assembly selected, the power density is 34.2 kW/l; the number of fuel assemblies is 708, and the equivalent core diameter is 4.65 m. The uranium enrichment of the refuelling batch, at equilibrium, is 3.6 %, and the average fuel burn-up is 39 GWd/t in the event of 23 months operation cycles.

5.3.2.3 Fuel handling and transfer systems

The fuel handling and refuelling operations are quite similar to those in current BWR designs. After the containment dome and the reactor vessel head have been removed, the reactor pool above the vessel opening is filled with water to provide radiation shielding. Then the reactor internals are lifted out of the vessel, using the overhead crane, and placed in storage positions in the pool. The procedures are simplified, however, because the internals to be treated are only the steam dryers due to the elimination of the steam separators. The handling of the fuel is made by means of a traditional refuelling machine. Spent fuel assemblies are brought up to the reactor pool and transported to storage racks with a capacity of 5 operating cycles in an adjacent fuel storage pool.

5.3.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel is 23 m high and with an internal diameter of 6.3 m. The operating pressure is 7 MPa as in the ABWR, and the design pressure of the vessel is 8.7 MPa. The nozzles for the two main steam lines of 700 mm diameter and the two feedwater lines of 400 mm diameter are located in the upper portion of the vessel; no large diameter nozzles are situated below the top of the core.

Reactor internals

The reactor internals are simple. The core support structure and the arrangement of guide tubes for instrumentation systems and for the bottom-entry control rods are similar to those in other BWRs, but the natural, rather than forced, coolant recirculation results in somewhat reduced stresses. Above the core, there are more differences; there is a riser structure from the core outlet up to the steam dome to improve the natural circulation rate, and there are no steam separators.

Steam/water separation is achieved by gravity separation above the two-phase surface, but still steam dryers are installed at the top of the vessel for moisture separation, in order to ensure that the quality of the steam supplied to the turbine is acceptable.

Reactor recirculation pumps

The HSBWR is designed with natural circulation of the coolant and there are no recirculation pumps.

5.3.2.5 Reactor auxiliary systems

A reactor core isolation cooling (RCIC) system is installed to maintain the reactor water level by water injection from the condensate storage tank during isolation events. A reactor water clean-up (RWCU) system is installed to control the content of radioactive and corrosion products in the reactor water and to remove residual heat in case of failure of the residual heat removal (RHR) system. The number of RHR system trains is reduced to one by the system simplification with the accumulator and the common use of the reactor water clean-up system for residual heat removal.

Other auxiliary systems such as the spent fuel storage pool cooling system, are basically the same as in currently operating BWRs.

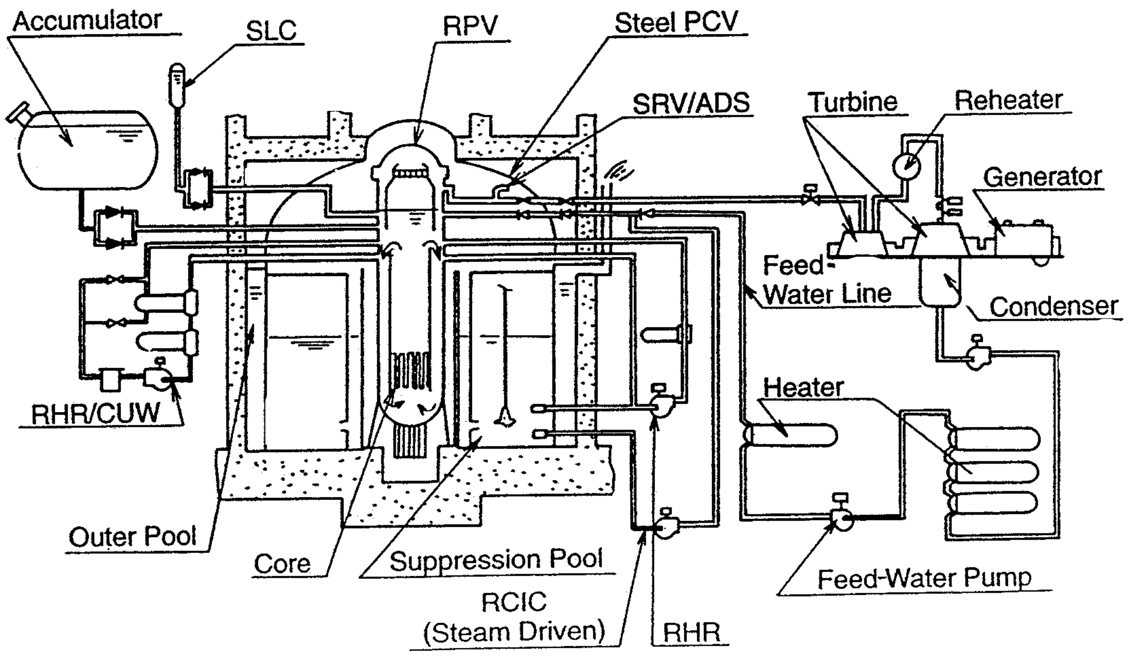


FIG. 5.3-2. HSBWR - System configuration.

5.3.2.6 Operating characteristics

The load following between 100% and 50% is mainly performed by control rod manoeuvering since there are no recirculation flow control possibilities. The harmonized control system utilizing Artificial Intelligence (AI) and control of the extraction steam for the turbine are taken into account.

5.3.3 Description of turbine generator plant systems

5.3.3.1 Turbine generator plant

The balance of plant of the HSBWR is mainly composed of two main steam lines, two feedwater lines, and the turbine systems, including one high pressure turbine and three low pressure turbines, etc.

The main turbine is a tandem compund, double flow turbine with 52" (1320 mm) blades on the last stage of the low pressure (LP) turbine units. The turbine, TCDF-52, is an adoption of the TC6F-52 which had already been developed for the 1350 MW(e) BWR (ABWR), in order to simplify the turbine island. A moisture separation and steam reheating system of the same type as in the ABWR plant is installed to improve the efficiency.

Bypass system is not shown and described in the text.

5.3.3.2 Condensate and feedwater systems

The condensate transfer pump is minimized by using the re-entry type. A high efficiency of about 33.4% is estimated under the condenser pressure of 5.07kPa. The feedwater system consists of the main feed pump, feedwater heaters, and piping in the same way as in the currently operating BWRs, even though the number of heaters is reduced from six to four per train, and the number of trains has decreased to one.

5.3.3.3 Auxiliary systems

The auxiliary systems are basically the same as in currently operating BWRs.

5.3.4 Instrumentation and control systems

5.3.4.1 Design concepts, including control room

The concept of I&C and electrical systems in HSBWR are basically the same as those of the ABWR. However, the electrical load is much smaller since the recirculation system has been eliminated. Furthermore, I&C and electrical systems are much simplified, e.g. the capacity of the emergency diesel generators is reduced, due to the adoption of passive safety systems.

5.3.4.2 Reactor protection and other safety systems

The reactor protection system and the other safety-related control systems are basically the same as those in current BWRs. The signals for the reactor protection system are generated in 2-out-of-4 coincidence logics. The digitalization of control panels and monitoring with CRT (cathode ray tube) displays are the main differences from the currently operating BWRs.

5.3.5 Electrical systems

5.3.5.1 Operational power supply systems

The concept of the ordinary power distribution is basically the same as that of the current BWRs. The power supply system capacity is greatly reduced, however, due to the elimination of the recirculation pumps. The number of DC power supply trains in the ordinary system is reduced to one.

5.3.5.2 Safety-related systems

The power supply for the safety-related system is reduced in accordance with the introduction of passive safety systems. One diesel generator unit of 1500 kW is provided to improve power supply security; it is not needed for an emergency. The number of DC power supply trains in the safety-related system is two. The batteries have a capacity that is sufficient for 2 hours at rated load.

5.3.6 Safety concept

5.3.6.1 Safety requirements and design philosophy

The safety design philosphy are as follows:

- To increase reliability of systems and avoid system complexity by adoption of passive safety systems;
- To increase grace period with respect to severe accidents.

The safety requirements are that there shall be no core uncovery in the event of design basis accidents and containment heat removal shall be ensured for 3 days without external power or other external support functions. The first requirement is the same as for the ABWR, and the corresponding design requirement is that no large diameter piping shall be connected to the reactor pressure vessel below the top of the core has been taken into account in the vessel design. A relatively large capacity water accumulator is introduced for emergency core cooling (ECC) purposes instead of the systems with motor-driven ECC pumps in the currently operating BWRs. The second requirement is met by the introduction of a water-wall containment cooling system which takes advantage of the steel primary containment vessel structure.

5.3.6.2 Safety systems and features (active, passive and inherent)

The HSBWR configuration provides a high safety level with respect to postulated loss-of-coolant-accidents (LOCAs) by pipe breaks, because there are no large diameter pipes connected to the reactor pressure vessel below the top of the core. The safety systems have redundancy by the combination of active and passive systems.

The steam driven reactor core isolation cooling (RCIC) system is provided for loss of all AC power (i.e. station blackout) situations and are also effective for small break LOCAs. The automatic depressurization system (ADS) and accumulators with emergency coolant at a low pressure (some 0.5 - 1 MPa) provide for short term emergency core cooling, instead of the high and low pressure pumped injection systems and emergency diesel generators in the current BWR designs. Elimination of emergency diesel generators and pumped injection systems simplifies the emergency core cooling systems (ECCSs) and provides for a high system reliability because of fewer components. The ADS flow area, initial pressure, and piping diameters of the accumulators are determined to realize no initial core heat-up and no core uncovery in any LOCA situation. The accumulators have the capacity to keep the whole core covered with coolant during the initial 24 hours after occurrence of an accident.

Core cooling after normal reactor shutdown and long-term core cooling after reactor scram are performed by a residual heat removal (RHR) system with injection pumps and heat exchangers. The RHR system has the capability to cool down the reactor to 52°C within 20 hours, and its capacity is sufficient also for heat removal during accidents. The accumulators and the suppression pool can remove and absorb heat for one day after reactor scram by themselves. Even when the RHR system is not available, coolant can be fed into the RPV by manually refilling the accumulators with attachable pumps, and heat removal from the RHR is performed by heat transfer from the suppression pool through the steel-walled primary containment vessel (PCV) to the outer pool. This heat removal from the PCV to the outer pool needs no additional systems. The outer pool has the capacity to maintain heat removal from the PCV to the outer pool for more than three days. Also, it is possible to feed water into the outer pool when it is requested.

5.3.6.3 Severe accidents (Beyond design basis accidents)

With respect to severe accidents, the containment cooling capability without any operator procedures is under investigation. The containment cooling by the outer pool (a kind of water wall type passive containment cooling system) is maintained for more than three days without any procedures.

5.3.7 Plant layout

5.3.7.1 Buildings and structures, including plot plan

The layout of the reactor island is shown schematically in Figure 5.3-3. The volume of the reactor building is about 50% of a current BWR building for the same rated capacity, which is realized by simplifying the components and systems as described above and by moving the spent fuel pool and control room to other buildings. All water pools are manufactured to the high level of welding technology and skill available in Japan, and steel structures are adopted for the primary containment vessel (PCV). Simplification of components and systems and adoption of steel structures in the PCV make it possible to achieve a shortened construction period which will be 32 - 36 months from the start of construction to commercial operation.

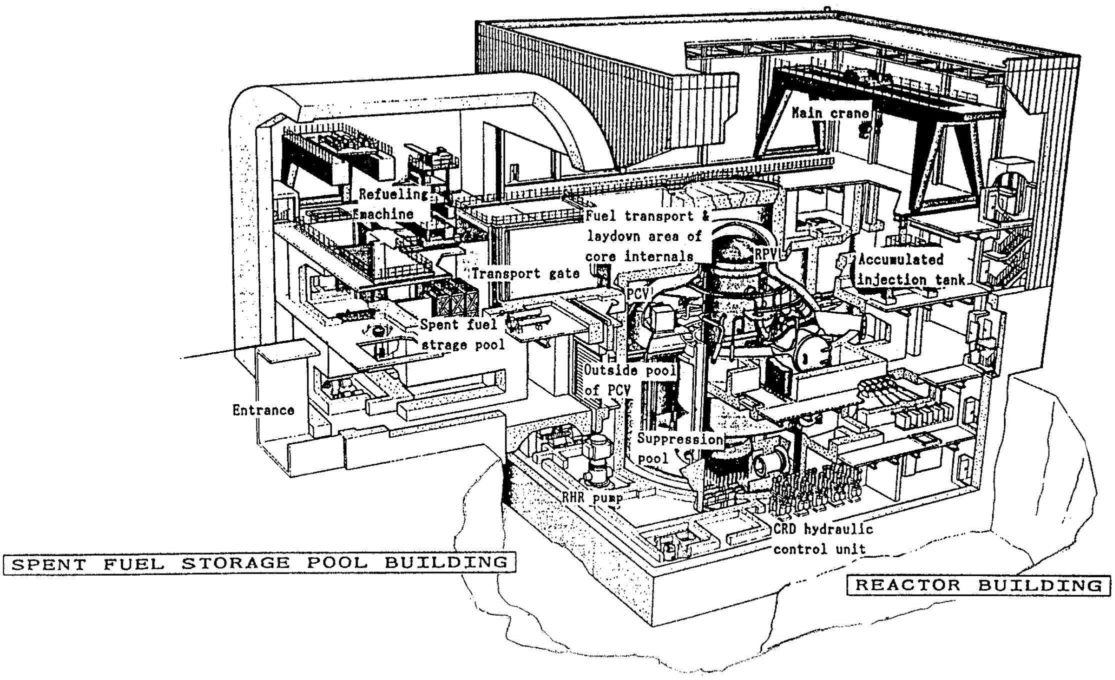


FIG. 5.3-3. General layout arrangement of the HSBWR reactor.

Design requirements

Earthquake: The seismic design is based on the Japanese guide S1 wave scaled to 0.35g for the firmest ground (with Vs+2000 m/s).

Aircraft crash, Explosion pressure wave, Internal hazards, Physical separation aspects, and Radiation protection aspects (accessibility, shielding, ventilation): No information provided

5.3.7.2 Reactor building

The reactor building encloses the containment and houses service systems for the reactor, such as handling equipment for fuel and main components, fuel pool, reactor water cleanup system and engineered safety systems.

In comparison with a current-day BWR of the same rated power, the building volume is decreased to 46% by the elimination of the recirculation pumps, ECCS pumps, and emergency diesel generators. The control room is separated and installed in another building to make the reactor building even more compact. The outer pool is located between the containment wall and the reactor building wall. The dimensions of the reactor building are 47 m width times 47 m depth times 47 m height. The seismic design is based on the Japanese guide S1 wave scaled to 0.35g for the firmest ground (with Vs+2000 m/s), and the standardization of the reactor building for any ground firmness is confirmed.

5.3.7.3 Containment

The containment is a free-standing steel containment vessel with a geometry that corresponds to the pressure suppression type containment of the current BWRs. The dimensions of the containment are 23 m diameter times 35 m height, and the building volume is about 70% of that of a current BWR of the same rated power.

Containment integrity, and corium retention

No information provided

5.3.7.4 Turbine building

The turbine building is directly adjacent to the reactor building enclosing the BOP systems such as the HP and LP turbines, the generator, the main condenser, and the feedwater pump. The dimensions of the turbine building are 47 m width times 58 m depth times 45 m height, and the volume is about 56% of that of a current BWR of the same rated power due to the system simplifications.

5.3.7.5 Other buildings

Each plant (unit) will be provided with an electrical/control building, a service building, and a fuel storage building, and there will be one waste building per four units at a power station. The structures of these buildings will be quite conventional.

5.3.8 Technical data

General plant data

Power plant output, gross	600	MW(e)
Power plant output, net		MW(e)
Reactor thermal output	1800	MWt
Power plant efficiency, net		%
Cooling water temperature	19	°C

Nuclear steam supply system

Number of coolant loops	1	
Primary circuit volume		m^3
Steam flow rate at nominal conditions	903.3	kg/s
Feedwater flow rate at nominal conditions	899.2	kg/s

Reactor coolant system

Primary coolant flow rate	4 944	kg/s
Reactor operating pressure	7.0	MPa
Steam temperature/pressure	289/7.0	°C/MPa
Feedwater temperature	185	°C
Core coolant inlet temperature		°C
Core coolant outlet temperature		°C
Mean temperature rise across core		°C

Reactor core

Active core height	3.1	m
Equivalent core diameter	4.65	m
Heat transfer surface in the core		m^2
Average linear heat rate		kW/m
Fuel weight		t U
Average fuel power density		kW/kg U
Average core power density	34.2	kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		
Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	3 700	mm
Rod array	8×8 in square lattice	
Number of fuel assemblies	708	
Number of fuel rods/assembly	60	
Number of spacers		
Enrichment (range) of first core, average		Wt%
Enrichment of reload fuel at equilibrium core	3.6	Wt%
Operating cycle length (fuel cycle length)	23	months
Average discharge burnup of fuel	39 000	MWd/t
Cladding tube material	Zircaloy	
Cladding tube wall thickness	0.86	mm
Outer diameter of fuel rods		mm
Fuel channel/box; material		
Overall weight of assembly, including box		kg
Uranium weight/assembly		kg
Active length of fuel rods	3 100	mm
Burnable absorber, strategy/material	Gd_2O_3 mixed with fuel	
Number of control rods	169	
Absorber material	B_4C/Hf	
Drive mechanism	electro-mechanical	
Positioning rate		mm/s
Soluble neutron absorber	not used	

Reactor pressure vessel

Inner diameter of cylindrical shell	6 300	mm
Wall thickness of cylindrical shell		mm
Total height, inside	24 550	mm
Base material: cylindrical shell	low-alloy carbon steel	
RPV head		
lining		
Design pressure/temperature	8.79/302	MPa/°C
Transport weight (lower part) [include. head]		t
RPV head		t

Reactor recirculation pump 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	Single wall, Pressure-suppression	
Overall form (spherical/cyl.)	cylindrical, steel	
Dimensions (diameter/height):	23/35	m
Design pressure/temperature	550/170	kPa/°C
Design leakage rate	0.5	vol%/day
Is secondary containment provided?	Yes, surrounding reactor building	

Reactor auxiliary systems

Reactor water cleanup,	capacity	22	kg/s
	filter type	deep-bed ?	
Residual heat removal,	at high pressure	250	kg/s
	at low pressure (100 °C)	6.3	MW
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage		kV
	rated capacity		MVA
Plant transformers,	rated voltage		kV
	rated capacity		MVA
Start-up transformer	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		2	
	rated power	1.5	MW
Number of diesel-backed busbar systems			
Voltage level of these			V AC
Number of DC distributions		0	
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) TCDF-52	Tandem compound, double flow	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/3 LP	
Turbine speed	1 500	rpm
Overall length of turbine unit	22	m
Overall width of turbine unit		m
HP inlet pressure/temperature	6.92/	MPa/°C

Generator

Type	4-pole, 3-phase, turbo-generator	
Rated power	670	MVA
Active power		MW
Voltage	20	kV
Frequency	60	Hz
Total generator mass, including exciter		t
Overall length of generator		m

Condenser

Type	shell type	
Number of tubes		
Heat transfer area	41290	m^2
Cooling water flow rate	41.9	m^3/s
Cooling water temperature	19	°C
Condenser pressure	5.07	kPa

Condensate pumps

Number	3 Hp & 3 LP	
Flow rate [1850 m^3/h]	500	kg/s
Pump head HP/LP [190/150 m WG]	1.86/1.47	MPa

Temperature		°C
Pump speed		rpm

Condensate clean-up system

Full flow/part flow	full condensate flow	
Filter type	rod type	

Feedwater tank

Volume		m^3
Pressure/temperature		MPa/ °C

Feedwater pumps

Number		
Flow rate		kg/s
Pump head		MPa
Feed pump power		MW
Feedwater temperature (final)	185.2	°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages,	low pressure	4
	high pressure	1
	feedwater tank	NA

5.3.9 Measures to enhance economy and maintainability

Measures taken to simplify the design include:

- Natural circulation for coolant recirculation;
- A water accumulator is adopted for emergency core cooling system (ECCS) instead of the systems with motor-driven ECCS pumps.

The cost of equipment and structures is reduced by eliminating forced recirculation systems and steam separators.

The construction period is reduced by simplification of components and adoption of a steel structure primary containment vessel. Cost and construction period have also been improved because it has been found possible to reduce the reactor building volume by more than 50% and the turbine building volume by more than 40%, as compared with the current BWR of the same rated power, through rationalization of the turbine and its associated equipment and simplification of the equipment relating to the nuclear steam supply system (NSSS) (i.e. recirculation pumps, ECCS pumps, and emergency diesel generators).

5.3.10 Project status and planned schedule

The HSBWR design is still at the conceptual design stage and licensing reviews have not yet been started.

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5.4 HABWR (HITACHI LTD., JAPAN)

5.4.1 Introduction

(1) Overview

Considering that energy demand is ever increasing all over the world, it is important to make nuclear power spread widely. Also, the recent electricity market in developed countries strongly requests minimized capital risks and timely return on an investment in consideration of the fact that the market is rapidly being liberalized. Having recognized these and other related circumstances, Hitachi is developing the HABWR (Hitachi Advanced Boiling Water Reactor) for the following major purposes:

- As a distributed power source that can be deployed in areas where an extensive transmission network can hardly be built and/or in areas where there are no or only weak transmission systems;
- As a power source that features flexibility to various market needs, such as minimized capital risks, timely return on a capital investment, etc.

The HABWR is a 600 MWe-class reactor which is considered the optimum scale for a power source which completely satisfies the targeted purposes mentioned above. The HABWR is so designed that it compensates for the loss of scale effect by best utilizing its inherent characteristics which accompany its medium plant power output. Also, it is intended that this be achieved without sacrificing the advantageous features of the latest ABWRs (Advanced Boiling Water Reactors). Since the HABWR is to be based on both the rich experiences in ABWR construction/operation in Japan and achieving design certification in the US, it is expected that many of its components and systems do not require special new developments.. This experience base should facilitate commercial introduction of the HABWR.

(2) Basic design policy

In countries, mainly in Southeast Asia, where energy demand is expected to expand rapidly, the need for medium-sized distributed power sources will increase because of limitations of investments, transmission networks, etc. Further, the markets in the various developed countries which have already introduced nuclear power have explicitly or implicitly showed the same or similar need for medium-sized distributed power sources. In order to meet this strong need from the market in the near term, it is important that the reactor be not only competitive in cost efficiency but also supported by already developed and matured technology proven by good operational results.

In the light of those key requirements, Hitachi has established the following basic design policy for the HABWR:

(a) Design that attains a targeted cost efficiency comparable to that of a large reactor while securing high levels of reactor safety, reliability, operability, and maintainability;

(b) Design with a high flexibility that can respond to specific local conditions, the capital strength, and customer needs; and

(c) Rational design that utilizes the technologies developed for larger reactors.

In addition to the above, as a design philosophy, superiority in regard to licensibility is also focused by employing proven and matured technologies. Thanks to several past experiences relating to the ABWRs, such as a design certification (DC) approved by the Nuclear Regulatory Commission (NRC) in the U.S., the actual construction and operating experiences including obtaining necessary licenses in

Japan, it is expected that the HABWR does not require a long-term licensing process which would be accompanied by considerable costs. This further reinforces advantages of the HABWR in terms of minimizing capital risks by shortening a required lead-time for an actual construction.

(3) Main plant system configuration

The general plant system configuration based on the above basic design policy is shown in Figure 5.4.1-1. The salient characteristics of the plant are as follows:

(a) Rational design considering trade-offs between plant performance and cost efficiency

As the plant power output is reduced, the thermal efficiency decreases accordingly. In developing the HABWR, despite the fact mentioned above, Hitachi decided to place design emphasis on thermal efficiency improvement in principle. In addition, simplification of system configuration and components was the other aspect of the design goal of the HABWR. From this standpoint, a turbine with 52” (about 132 cm) blades which was developed for the ABWR (1350 MWe class) is employed for the HABWR in place of the two low-pressure turbines adopted for BWRs with comparable electrical output. In addition, the HABWR employs single-shell type condensers and only one string of feedwater heaters. This system configuration should allow for significant equipment rationalization, although the thermal efficiency is slightly lower than that of the ABWR..

(b) Simplification of system configuration

Concerning the safety systems too, they have been simplified without sacrificing the performance of the existing systems. The components of the emergency core cooling systems (ECCSs) require development of environmental tests, etc. because of their severe operating conditions. Therefore, for certain singular components, e.g. ECCS, etc., the capacity of each of them follows that of the ABWR (1,350 MWe class) to eliminate the need to redesign them. Also, the system configuration is optimized to allow for significant simplification of the equipment (number of components reduced to approximately one-half) while securing the same level of safety as the existing systems.

(c) Utilization of technologies developed for larger reactors

The new technologies established in the development of larger reactors, such as the large-capacity safety relief valve (SRV) and the low-pressure-loss main steam isolation valve (MSIV), are implemented in the design to improve the plant performance.

(d) Rationalization of layout design

It has been found possible to reduce the building volume by more than 50%, as compared with the ABWR (1,350 MWe class), through rationalization of the turbine and its associated equipment and simplification of the equipment relating to the nuclear steam supply system (NSSS).

In order to attain a targeted cost efficiency (unit cost of construction) comparable to that of the ABWR (1,350 MWe class), Hitachi intends to efficiently implement an extended use of the modularization technology by thorough standardization of the equipment specifications and adoption of state-of-the-art technologies that are being put into practical use, such as seismic isolation technology and SC (steel plate reinforced concrete) construction, etc. It is believed that this will facilitate introducing an HABWR of optimum plant power output according to power demand and investment scale of each market.

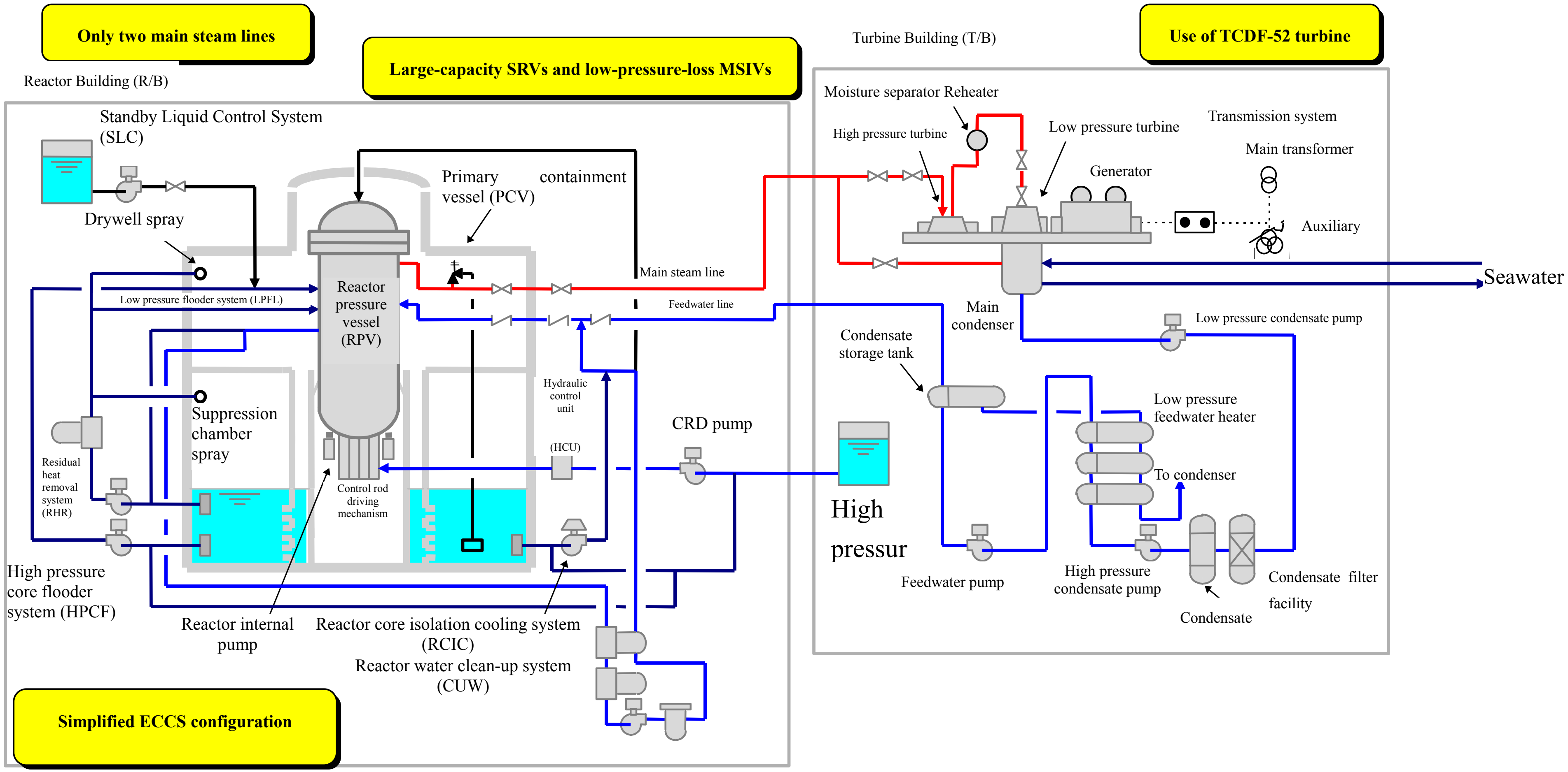

FIG. 5.4-1. *Scheme of HABWR plant system configuration*

5.4.2 Description of the nuclear systems

5.4.2.1 Primary circuit and its main characteristics

The cross section of the reactor pressure vessel is shown in Figure 5.4-1.

For coolant recirculation, the forced circulation system whose performance has been proved by the ABWR is adopted. There are four reactor internal pumps (RIPs) of built-in type.

The HABWR has two main steam lines, each being 700 mm in diameter. Each line is equipped with two MSIVs of low pressure loss. In addition, a total of seven large-capacity SRVs are provided for these two main steam lines.

5.4.2.2 Reactor core and fuel design

The core and reactor are designed as compact as possible for the predetermined plant power output (about 600 MWe). The thermal output required for 600 MWe is about 1,700 MWt or more and the corresponding number of fuel assemblies is about 340. Since it is possible to increase the number of fuel assemblies up to 376 and still secure the required output without changing the size of the pressure vessel and the numbers of control rods (CRs) / control rod drive mechanisms (CRDs), the number of fuel assemblies and the thermal output were decided to be 376 and 1,862 MWt, respectively.

In addition, the RIP specifications were reviewed on the basis of the assumed core flow rate. As a result, it was found that the required core flow rate could be secured by four RIPs if their operating range was extended to their maximum capacity.

5.4.2.3 Fuel handling and transfer systems

The fuel handling and refueling operations are quite similar to those in current BWR design.

5.4.2.4 Primary components

Reactor pressure vessel (RPV)

The RPV is 20 m in height and 5.5 m in inside diameter. These dimensions are required by the number of fuel assemblies and the height of the core, the height of dryers and separators, etc. As in the case of the ABWR, the operating pressure and the maximum design pressure are 7 MPa and 8.7 MPa, respectively. The nozzles for the two main steam (MS) lines are 700 mm in diameter, the two feedwater (FDW) lines are 400 mm in diameter, and the four riser tubes are 250 mm in diameter.

Reactor internals

As in the ABWR, dryers and separators are installed inside the RPV.

5.4.2.5 Reactor auxiliary systems

The system specifications are outlined below.

Residual heat removal (RHR) system

As in the ABWR, the RHR system has the functions of removing the decay heat after reactor shutdown, injecting water into the reactor during a loss-of-coolant accident (LOCA), and cooling water of the suppression pool (S/P). There are two RHR systems, each having 100% capacity. Both systems are so configured that they are capable of cooling the S/P water independently.

Reactor core isolation cooling (RCIC) system

As in the ABWR, the RCIC system has the functions of feeding water to the reactor for cooling it during reactor isolation and injecting water into the reactor during a LOCA. The HABWR is provided with one RCIC system.

High-pressure core flooder (HPCF) system

As in the ABWR, the HPCF system has the function of injecting water into the reactor during a LOCA. In the existing ABWR, two HPCF systems are equipped to reinforce the ECCS. In the HABWR, however, only one HPCF system is sufficient for the required capacity. Therefore, the number of HPCF systems has been reduced to one. It should be noted, however, that the function of the automatic depressurization system (ADS) has been enhanced, and the function of the high-pressure ECCS has been reinforced by implementing an increase of the RHR pump head.

Clean-up Water (CUW) system

As in the ABWR, the CUW system has the functions of removing corrosion products, fission products, etc. from the reactor coolant during normal operation and shutdown, and controlling the level of reactor water during reactor start-up, etc. The HABWR is provided with one CUW system. The system flow rate is about 2% of the main feedwater flow rate as in the ABWR. A heat exchanger for larger reactor is employed to reduce the number of CUW non-regenerative heat exchangers, thereby rationalizing the equipment and layout space.

Fuel pool cooling and filtering (FPC) system

As in the ABWR, the FPC system has the function of removing the decay heat generated by spent nuclear fuel and the corrosion products, fission products, and soluble inorganic material carried in by spent nuclear fuel. The HABWR is provided with two FPC systems, each having 100% capacity for the cooling function and 50% capacity for the clean-up function.

Reactor cooling water (RCW) system/reactor seawater (RSW) system

As in the ABWR, the RCW/RSW systems have the functions of cooling the equipment during normal reactor operation, removing the residual heat from the reactor in an emergency, and cooling the safety-related equipment. The HABWR is provided with two RCW systems and two RSW systems. In addition, one RCW system and one corresponding RSW system are provided for cooling the emergency diesel generator (DG) exclusive for the HPCF.

Emergency diesel generator (DG)

As in the ABWR, the DGs have the function of supplying power to the designated equipment during a loss of power accident (LOPA) or a LOCA with LOPA. The HABWR is provided with two DG systems. In addition, one DG system exclusive for the HPCF is provided.

Control rod drive mechanism (CRD)

As in the ABWR, the CRD system employs a fine-motion control rod drive mechanism (FMCRD) actuated by a motor and water pressure.

5.4.2.6 Operating characteristics

The operating characteristics of the HABWR are almost the same as those of the existing ABWR since the plant system configuration is basically the same as that of the existing ABWR.

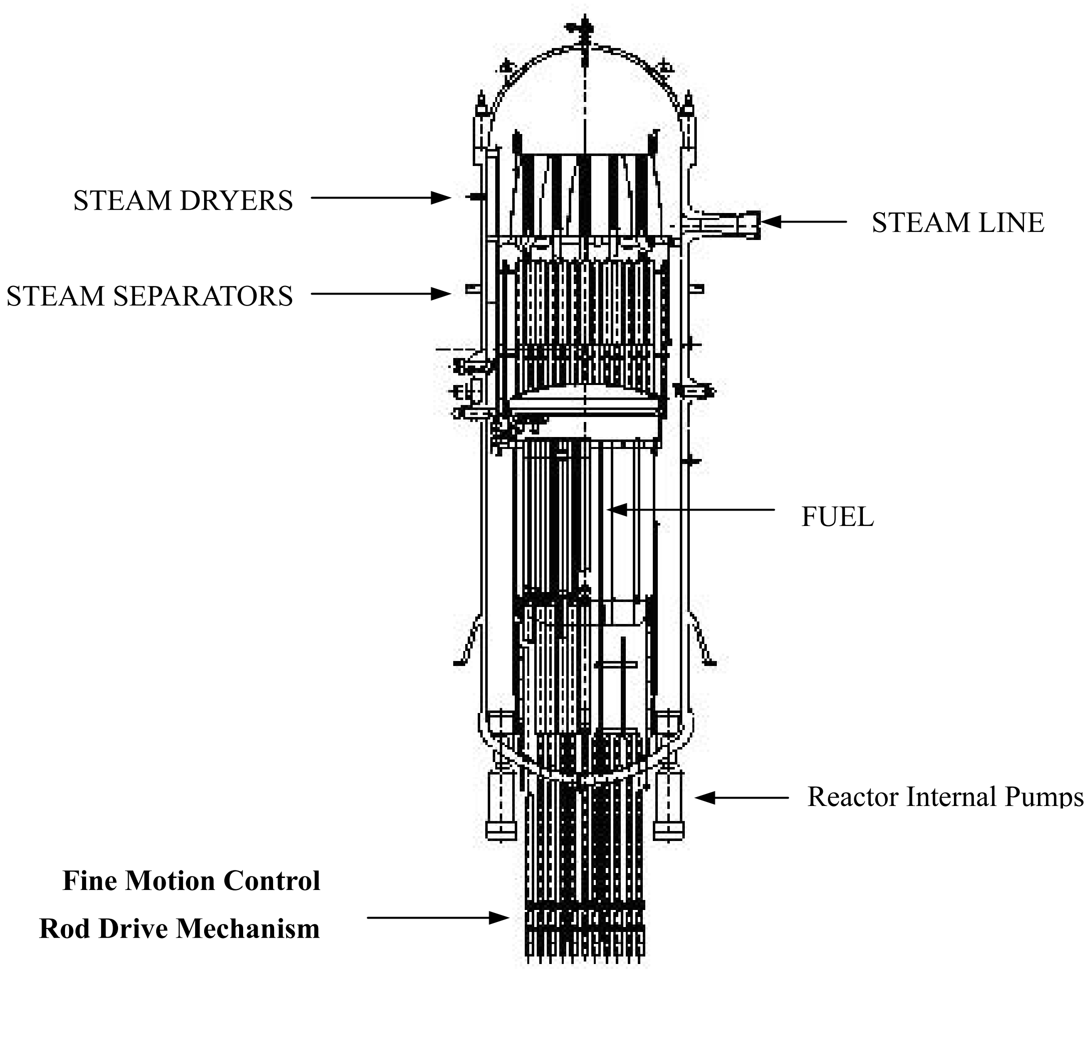

FIG. 5.4-2. Cross section of reactor pressure vessel

5.4.3 Description of the turbine generator plant systems

The main BOP systems of the HABWR consist of two main steam lines, two feedwater lines, high-pressure (HP) and low-pressure (LP) turbine systems, etc. Basically, the components and systems were decided with major design emphasis placed on obtaining high thermal efficiency of the turbine system.

5.4.3.1 Turbine generator plant

Turbine type

The type of main steam turbine is determined by the combination of the length of last-stage turbine blade and the number of exhaust flows. In consideration of the thermal efficiency and equipment cost of turbines, the turbine type is a tandem-compound, dual-flow (TCDF) turbine with 52” blades. One high pressure turbine and one low pressure turbine are installed.

Turbine thermal cycle

Generally, the turbine thermal cycles employed in nuclear power plants in Japan are either the two-stage reheating system or the non-reheating system. In recent years, many nuclear power plants employ a reheating system with a moisture separator reheater to improve the thermal efficiency. Therefore, the two-stage reheating system is adopted for the HABWR.

5.4.3.2 Condensate and feedwater systems

Numbers of feedwater heater stages and systems

The number of feedwater heater stages influences plant power output, i.e. thermal efficiency, and equipment cost. In the HABWR, the number of heater stages is 6 (2 stages for high pressure, 4 stages for low pressure) as in the ABWR. In consideration of the number of turbine casings, the feedwater heaters are equipped in one system with a total of six stages.

Feedwater heating cycle

There are two methods for treating drain from the feedwater heaters: the drain cascade method (the entire amount of drain is returned to the condenser) and the drain pump-up method (the drain is recycled directly to the condensate system). The HABWR adopts the drain pump-up system as does the existing ABWR.

Feedwater pump driving mechanism

In view of major type of feedwater pump driving mechanism, there are turbine driven mechanism and motor driven mechanism. At large-capacity power plants, the turbine driven type is employed from the viewpoint of thermal and cost efficiency, whereas the motor driven type is generally used at small-capacity power plants. For the HABWR of 600 MWe class, therefore, a motor driven type is adopted.

5.4.3.3 Auxiliary systems

The auxiliary systems are basically the same as those of currently operating BWRs/ABWRs.

5.4.4 Instrumentation and control (I&C) systems

Basically, a high-reliability design comparable to that of ABWRs is adopted for the HABWR. That is, integrated digitalized system is applied in the I&C of HABWR. It should be noted, however, that some of the instrumentation and control systems have been simplified and sophisticated in consideration of its medium power output and simplified plant equipment. The contents of the system simplification and sophistication are described below.

(1) Reduction of size of large display panel

The operating units (flat display, hard switches, etc.) installed on the large display panel are removed and only the large display unit shall be installed as an equipment classified in a lower grade of seismic design category, e.g. non-category 1 in the U.S., class C in Japan, etc. As a result, the large display panel can be reduced in size. The operating units removed from the large display panel are relocated either on an expanded area of the main console or on the maintenance operating console.

(2) Integration of control devices of non-safety systems

In view of the sophistication of functions and the increase in processing speed of control devices, control devices of the main control systems are shared among each system as much as possible. The main control systems (the feedwater control system (FDWC), the recirculation flow control system (RFC), the automatic power regulator system (APR), etc.) have been composed of triple control devices to maintain a high reliability. With the triple control concept kept unchanged, the control devices of the main control systems are shared among each system as much as possible to simplify the equipment and save the installation space.

For other non-safety control systems, in the existing technologies, dual control devices are provided to maintain a high reliability and enhance the functions of monitoring and operating the controlled devices during periodical inspection, etc. With this design concept of dual control devices kept unchanged, control devices of several non-safety control systems are integrated to simplify the control equipment and reduce the installation space to approximately one-half in digital control systems. This integration can be implemented as on-line maintenance of dual control devices becomes possible.

(3) Adoption of efficient and sophisticated communication technology in the field

In the ABWRs, multiplex signal transmission technology is extensively applied in communications between the field devices/instruments and the monitor and control equipment installed in the central control room so that the signal cables can be reduced by multiplex transmission of field signals. For the HABWR, these communication technologies are made more efficient and more sophisticated to allow for integrated processing of field signals. For example, concerning the conventional power supply panel that is provided with a signal I/O panel installed separately, a compact signal I/O device is incorporated in the power supply panel to eliminate the need to install the signal I/O panel. This not only eliminates the signal I/O panel but also reduces the amount of control cables between the panels.

5.4.4.1 Design concept, including control room

The description of major design features of the HABWR included in this category is outlined within 5.4.4. The other design features except those mentioned in 5.4.4 are basically the same as those of the ABWR.

5.4.4.2 Reactor protection and other safety systems

The description of major design features of the HABWR included in this category is outlined within 5.4.4. The other design features except those mentioned in 5.4.4 are basically the same as those of the ABWR.

5.4.5 Electrical systems

Basically, a high-reliability design comparable to that of the ABWRs is adopted for the electrical systems of the HABWR. It should be noted, however, that some of the electrical systems have been simplified/sophisticated in consideration of its medium power output and simplified plant equipment. The contents of the simplification/sophistication are described below.

(1) Single auxiliary transformer, 2-bus metalclad switchgears (M/Cs) of normal, and 40 kA interrupting current of circuit breaker

Since the HABWR has a lower power output than ABWRs and the station auxiliary loads capacity within the plant are reduced, the number of auxiliary transformers supplying those auxiliary devices and the number of metal clad (high-voltage) switchgear (M/C) are evaluated. Based on the evaluation results, the number of auxiliary transformers is reduced to 1 and the number of M/C of normal is reduced to 2. (while for the ABWR (1,356 MWe class), Auxiliary Transformer: 2, M/C of normal: 4)

(2) Elimination of stand-by transformer

For synchronizing the generator with the power system, the options are: the high-voltage synchronizing system, which uses a circuit breaker installed at the high-voltage side of the main transformer; and the low-voltage synchronizing system, which uses a generator load switch (GLS) installed at the generator output side (low-voltage side of the main transformer). The HABWR adopts the low-voltage synchronizing system.

By adopting the low-voltage synchronizing system, it is possible to eliminate the start-up transformer for securing the auxiliary power supply, etc. when starting the plant. However, a stand-by transformer is installed to provide electrical power against an electrical accident with the main circuit of the main transformer, etc. and for maintenance of the main circuit.

In the case of the HABWR, a stand-by transformer of large capacity is not equipped on the premise that the feedwater/condensate systems are not operated at an electrical accident, as in a reactor scram and/or a turbine trip, and that a small-capacity transformer is installed for maintenance of the main circuits.

(3) Rationalization of 250 V batteries for the turbine emergency oil pump (EOP)

At the existing nuclear power plants, emergency DC motors, for example, turbine emergency oil pump (EOP) are installed to prevent the turbines, generators, and other important devices from being damaged by a loss of AC power supply accident. The power sources of these are DC power supplies so that they can be operated even in the case that the AC power supply is interrupted. However, compared with the induction motor, the DC motor has a complicated construction and is inferior in maintainability.

On the other hand, an inverter is capable of stably supplying the load even in a loss of the AC power supply accident. Because of this, it is used as a power supply for instrumentation/control at nuclear power plants.

In view of the above facts, the HABWR adopts a sophisticated EOP power supply system with an induction motor for driving the EOP and an inverter for driving the induction motor.

(4) Reduction of capacity of emergency diesel generator (DG)

In the HABWR, the ECCS components that are main loads of the DG are made simpler than that of the ABWRs. Therefore, the capacity of the DG was reviewed. As a result, the electrical system configuration of the DG has been changed from 5,300 kW x 3 (for the ABWR (1,350 MWe class)) to 3,000 kW x 3 (capacity reduction of about 40%).

In addition, the following electrical system has been simplified/sophisticated:

- Single bus/division of the emergency power centers (P/Cs);
- Adoption of stationary valve regulated lead acid battery to DC system of safety.

5.4.5.1 Operational power supply system

The description of major design features of the HABWR included in this category is outlined within 5.4.5. The other design features except those mentioned in 5.4.5 are basically the same as those of the ABWR.

5.4.5.2 Safety-related systems

Description of major design features of the HABWR included in this category is outlined within 5.4.5. The other design features except those mentioned in 5.4.5 are basically the same as those of the ABWR.

5.4.6 Safety concept

5.4.6.1 Safety requirements and design philosophy

The level of safety of the HABWR is the same as that of the ABWR; provided, however, that the levels of reliability, operability, and maintainability are those which are appropriate to the HABWR.

5.4.6.2 Safety systems and features

Safety equipment configuration

The major safety systems/components of ABWR that have been modified for the HABWR are described below:

(a) Main steam safety relief valve (SRV) and automatic depressurization system (ADS)

The SRV and ADS are of the same type as those of the ABWR. For the SRVs (7 valves) of the HABWR, a large-capacity SRV which has been developed for larger reactors is employed to reduce the number of components. In addition, the ADS function has been enhanced and the ADS function has been imparted to six of the SRVs to rationalize the high-pressure ECCS.

(b) Emergency core cooling system (ECCS)

The ECCS consists of one high-pressure core flooding (HPCF) system, two low-pressure core flooding systems (LPFLs), and one RCIC system. By enhancing the ADS functions, one of the high-pressure ECCSs has been eliminated compared to the ABWR. The ECCS configuration is schematically shown in Figure 5.4-3.

Safety performance evaluation

(a) ECCS performance evaluation

By using the evaluation techniques applied to the ABWRs, the ECCS network for the HABWR was analyzed taking into consideration the capacity of the HABWR, the reduction of RIPs to four units, the change of dimensions of the reactor pressure vessel, etc. As in the case of the ABWR, the assumed event is break of the HPCF piping that produces the severest result. The behavior of reactor water level is shown in Figure 5.4-4. From the results, when the reactor water level reaches L1, the ADS is actuated, depressurizing the reactor. At the same time, water is injected into the reactor from the LPFL, restoring the water level in the reactor. As a result, it was confirmed that the reactor water level is always kept above the top of the fuel rods and core coverage is maintained.

(b) Primary Containment Vessel (PCV) performance analysis

By using the evaluation techniques applied to the ABWRs, the PCV performance was analyzed taking into consideration for the change of diameter of the breaking part, the change of PCV dimensions, the change of number of vent tubes, etc. As in the case of ABWRs, the assumed event is a break of the feedwater piping that produces the severest result. The behavior of PCV pressure is shown in Figure 5.4-5. From the results, it was confirmed that the rise in PCV pressure is kept at nearly the same level as in the ABWR.

5.4.6.3 Severe accidents (beyond design basis accidents)

The safety performance of the HABWR to severe accidents is the same as that of the existing ABWR.

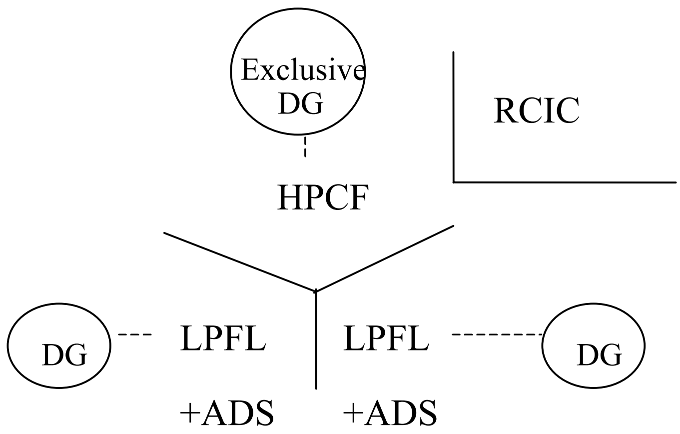

FIG. 5.4-3. ECCS network configuration

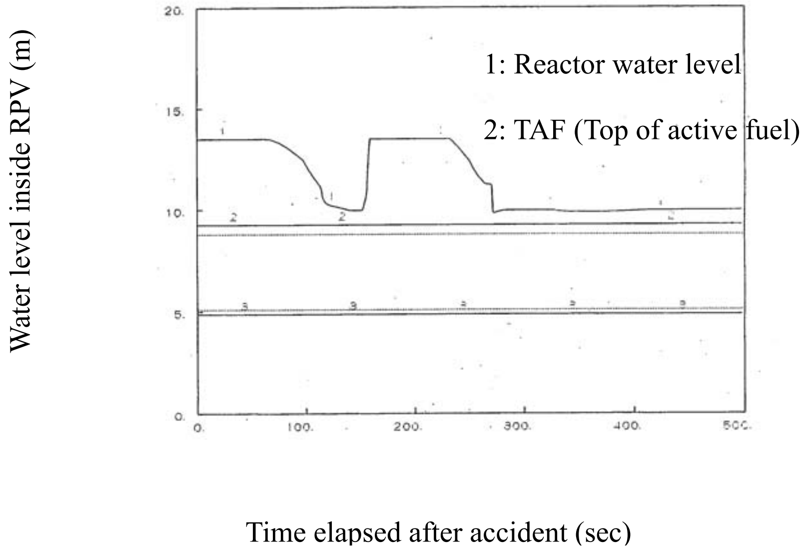

FIG. 5.4-4. Behavior of reactor water level (during HPCF piping break)

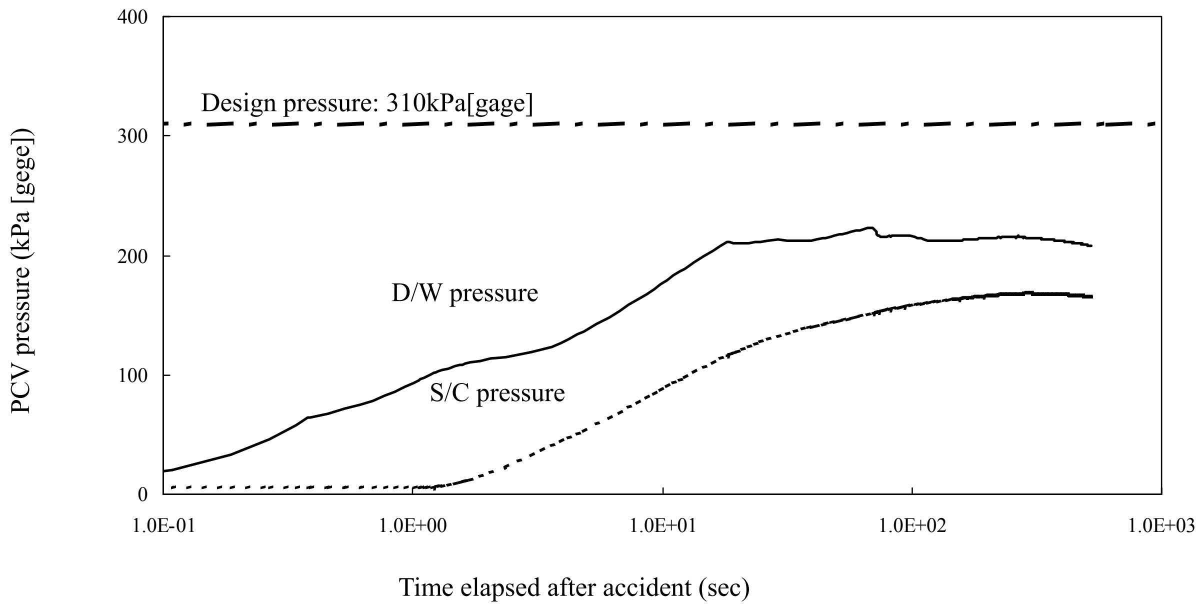

FIG. 5.4-5. Behavior of PCV pressure (during feedwater piping break)

5.4.7 Plant layout

5.4.7.1 Buildings and structures, including plot plan

The design concept of HABWR layout is based on design combinations of the proven ABWR technologies and existing technologies. Through the rationalization of equipment organization and the compact layout study, the HABWR has attained about 50% reduction of the building volume and the number of main system components as compared with the ABWR (1,350 MWe class). Those approaches achieve the construction period of around 34 months for the first unit and 31.5 months for the following units, respectively[1] by utilizing advanced construction technologies.

The key features of the layout of buildings for the HABWR are:

a. Compact layout: Reduction of building sizes and direct material quantity including civil structure by concentrated layout of facilities, as well as rationalization of equipment;

b. Standardization: Completely duplicated layout design for main islands corresponding to the system specifications;

c. Rapid construction: Reduction of building stories, consideration of equipment location which structures a critical path of construction process, and the expansion of area of high efficiency modularization.

In particular, in order to attain high level standardization, the islands are divided into fixed layout areas and variable layout areas, and the facilities to be housed in the reactor building and turbine building are limited so as to minimize the influences of the site conditions and other variable factors. Seismic isolation concept is also provided as an option in order to make further reduction of the components and completely duplicated design possible from a standpoint of seismic design for equipment and structure.

A Schematic bird's-eye/cutaway view of the HABWR is shown in Figure 5.4-6.

5.4.7.2 Reactor building

The reactor building accommodates the reactor containment vessel, the ECCS equipment, spent fuel pool, fuel handling equipment, the CUW equipment, safety-related equipment, etc. As a fixed layout area, the reactor building is laid out in a way to allow for standardization. Because of this, the DGs, the RIP power supply equipment, etc. are housed in a separate electrical equipment building. The location of electrical equipment which is related to the first system pre-operation test has been lowered and the number of building stories have been minimized to shorten the construction period.

5.4.7.3 Containment

The reactor containment vessel is based on the reinforced concrete containment vessel (RCCV) of the ABWR (1,350 MWe class). Reflecting the reduction in number of main steam pipes along with consideration of the thermal output level, the diameter and height of the containment vessel was carefully reviewed from a safety-design point of view and have been reduced.

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[1] From first concrete to fuel loading

5.4.7.4 Turbine building

The turbine building that adjoins the reactor building houses the HP/LP turbines, generators, condensers, feed-water pumps, and other turbine related equipment. The length of the long side of the turbine building has been reduced mainly by the application of TCDF turbine, installing the moisture separator re-heater on the operating floor, and rationalizing the equipment of the turbine systems. In addition, the equipment is laid out in a concentrated manner to reduce the main piping and the number of building stories has been reduced to facilitate building construction.

5.4.7.5 Other buildings

The electrical equipment building is an independent, low-storied building accommodating the central control equipment, the DGs, the RIP power supply equipment, non-safety electrical equipment, etc. This building is compact in size, and the building configuration and equipment layout are such that they do not constitute a critical path in the construction work.

The seawater heat exchanger building is also an independent building which houses the circulating water pumps, safety/non-safety seawater pumps, heat exchangers, etc. It is a variable layout area which can flexibly respond to the site conditions, for example water intake and discharge directions, and other variable factors.

FIG. 5.4-6. Schematic Bird's-eye/cutaway view of HABWR

5.4.8 Technical data

General plant data

Power plant output, gross	650	MWe
Power plant output, net		MWe
Reactor thermal output	1862	MWt
Power plant efficiency, net		%
Cooling water temperature		℃

Nuclear steam supply system

Number of coolant loops	1	
Primary circuit volume, including pressuriser		m3
Steam flow rate at normal conditions	1007	kg/s
Feedwater flow rate at nominal conditions	1005	kg/s
Steam temperature/pressure	288/7.17	℃/MPa
Feedwater temperature/pressure	216/7.17	℃/Mpa

Rector coolant system

Primary coolant flow rate	7111	kg/s
Reactor operating pressure	7.17	MPa
Coolant inlet temperature, at RPV inlet	278	℃
Coolant outlet temperature, at RPV outlet	288	℃
Mean temperature rise across core	10	℃

Reactor core

Active core height	3.7	m
Equivalent core diameter	3.4	m
Heat transfer surface in the core	3460.8	m2
Fuel inventory	64.7	tU
Average linear heat rate	5.8	kW/m
Average fuel power density	28.8	kW/kgU
Average core power density (volumetric)	55.6	kW/l
Thermal heat flux, Fq	538.0	kW/m2
Enthalpy rise, FH	53.3	kJ/kg
Fuel material	UO2	
Fuel assembly total length		mm
Rod array	9x9	
Number of fuel assemblies	376	
Number of fuel rods/assembly	74	
Number of control rod guide tubes		
Number of spacers	7	
Enrichment (range) of first core		Wt%
Enrichment of reload fuel at equilibrium core	4.3	Wt%
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel	45	MWd/t
Cladding tube material	Zy-2	
Cladding tube wall thickness	0.7	mm
Outer diameter of fuel rods	11.2	mm
Overall weight of assembly		kg
Active length of fuel rods		mm
Burnable absorber, strategy/material		
Number of control rods		
Absorber rods per control assembly		
Absorber material		
Drive mechanism	electro-mechanical/hydraulic	
Positioning rate		
Soluble neutron absorber		

Reactor pressure vessel

Cylindrical			
Shell inner diameter		5.3	m
Wall thickness of cylindrical shell			m
Total height		20	m
Base material:	cylindrical shell	low alloy steel	
	RPV head	low alloy steel	
	Liner	SUS	
Design pressure/temperature		8.62MPa/302℃	
Transport weight (lower part), including head			
	RPV head		

Steam generators — Not Applicable

Type

Number		
Heat transfer surface		m2
Number of heat exchanger tubes		
Tube dimensions		mm
Maximum outer diameter		mm
Total height		mm
Transport weight		t
Shell and tube sheet material		
Tube material		

<u>Reactor recirculation pump</u>

Type	variable speed, wet motor, single stage, vertical internal pump	
Number	4	
Design pressure/temperature	same as for RPV	MPa/℃
Design flow rate (at operating conditions)		kg/s
Pump head		m WG
Pump casing material	same as for RPV	
Pump speed		rpm

<u>Pressuriser</u> Not Applicable

Total volume		m3
Steam volume: full power/zero power		m3
Design pressure/temperature		MPa/℃
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter		mm
Total height		mm
Material		
Transport weight		t

<u>Pressuriser relief tank</u> Not Applicable

Total volume		m3
Design pressure/temperature		MPa/℃
Inner diameter (vessel)		mm
Total height		mm
Material		

Transport weight		t

<u>Primary containment</u>

Type	Pressure-suppression /reinforced concrete	
Overall form (spherical/cyl.)	cylindrical	
Dimensions(diameter/height)	23/35	m
Free volume(drywell/wetwell/cod.pool)		m3
Design pressure/temperature (DBEs)	310/171	kPa/℃
(severe accident situations)		kPa/℃
Design leakage rate	0.4vol%/day	
Is secondary containment provided?	Yes	

<u>Reactor auxiliary systems</u>

Reactor water cleanup, capacity		kg/s
Filter type		
Residual heat removal, at high pressure		kg/s
at low pressure		kg/s
Coolant injection at high pressure		kg/s
at low pressure		kg/s

<u>Power supply systems</u>

Main transformer, rated voltage		kV
rated capacity		MVA
Plant transformers, rated voltage		kV
rated capacity		MVA
Start-up transformer rated voltage		kV
Rated capacity		MVA
Medium voltage busbars		kV
Number of low voltage busbar systems		V
Standby diesel generating units: number		
rated power		MW
Number of diesel-backed busbar systems		
Voltage level of these		V ac
Number of DC distributions		
Voltage level of these		V ac
Number of battery-backed busbar systems		

Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	TCDF-52	
Number of turbine sections per unit (e.g. HP/LP/LP)	1HP / 1LP	
Turbine speed		rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/℃

Generator

Type		
Rated power		MVA
Active power		MW
Voltage		kV
Frequency		Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type	Single-shell type	
Number of tubes		
Heat transfer area		m2
Cooling water flow rate		m3/s
Cooling water temperature		℃
Condenser pressure		hPa

Condensate pumps

Number		
Flow rate		kg/s
Pump head		MPa
Temperature		℃
Pump speed		rpm
Containment vessel cooling systems	2 lines (100% × 2)	

	Containment vessel spray × 2 systems (shared with LPFL)
Emergency DG	2 units + 1 unit exclusive for HPCF
Reactor water cooling/seawater cooling systems	2 systems + HPCW/HPSW
Residual heat removal system	2 systems
Reactor core isolation cooling system	1 system
Clean-up water system	High pressure type; capacity: 2% of feedwater
Fuel pool cooling and filtering system	100% × 2 for cooling, 50% × 2 for filtering

BOP systems

Turbine type	TCDF-52 (high pressure: 1 casing, low pressure: 1 casing)
Heater for moisture separation	2-stage reheater
Condenser	Single-shell type
Heater configuration	High pressure × 2 stages/1 system
	Low pressure × 4 stages/1 system
Heater drain system	Drain pump-up type
Condensate system	Low pressure condensate pump × 2 (50% × 2)
	High pressure condensate pump × 2 (50% × 2)
Feedwater system	Motor-driven feedwater pump × 2 (50% × 2)

5.4.9 Measures to enhance economy and maintainability

- To simplify the design,

    - Only two main steam lines
    - Single-shell type condensers and only one system of feedwater heaters
    - Simplified ECCS (two series of RHR, one series of HPCF)
    - Elimination of stand-by transformer

- To reduce the cost of equipment and structures,

    - Large-capacity SRVs

- To reduce the construction period,

    - An extended use of the modularization technology by thorough standardization of the equipment specifications;

    - It has been found possible to reduce the building volume by more than 50%, as compared with the ABWR (1,350 MWe class), through rationalization of the turbine and its associated equipment and simplification of the equipment relating to the nuclear steam supply system (NSSS).

5.4.10 Project status and planned schedule

As mentioned above, the HABWR emphasizes rationalized/optimized design in terms of not only functionality, but cost efficiency. It is also aimed that this design philosophy is supported by well-proven/matured technologies/designs that have been employed in the past and existing BWRs/ABWRs. It is expected, therefore, that the HABWR will be materialized in the near future when its targeted cost efficiency, that is a unit cost of construction comparable to that of the ABWR (1350 MWe class), is attained.

It is also expected that the HABWR does not require a long-term licensing process in the light of its inherent licensibility thanks to several past experiences relative to the ABWRs, e.g. a design certification (DC) approved by the NRC in the U.S., and the actual construction and operating experiences including obtaining necessary licenses in Japan, etc.This also supports a possibility of a near-term deployment of the HABWR in an actual electricity grid.

References

MORIYA, K., et al., "Development Study of Nuclear Power Plants for 21st Century, The HITACHI Review, No.3, Vol.50, pp61-67 (2001).

5.5 WWER-640/V-407 (ATOMENERGOPROJECT/GIDROPRESS, RUSSIAN FEDERATION)

5.5.1 Introduction

Ensuring the safety of the personnel, the population and the environment against radiation effects is used as the basis for the design. The prescribed doses of exposure, and the standards for the release of radioactive substances and their content in the environment, should not be exceeded under normal operation, anticipated operational occurrences, and in design and beyond-design-basis accidents during the 50-60 years' service life of the plant.

Operating limits for fuel cladding damage are as follows:

- Up to 0.1% of fuel rods with flaws of gas leak-tightness, and up to 0.01% of fuel rods with direct contact between fuel and coolant;
- 7.4E10 Bq/m^3 primary coolant iodine nuclide radioactivity;
- The iodine nuclide radioactivity in secondary side water of any steam generator should not exceed 1.5E4 Bq/m^3 under normal conditions and operational occurrences. The estimated probability of the operating limits being exceeded is less than 1E-2 per reactor-year.90

Fuel element damage leading to:

- considerable amount of radioactivity release from fuel rods,
- considerable steam-zirconium reaction progression (considerable from the standpoint of fulfillment of the safety insurance requirements mentioned above),
- fuel material escape out of the cladding preventing core cooling and post-accident removal,

are design limits that should not be exceeded in design basis accidents. The estimated probability of the design limits being exceeded must be less than 10^{-5} per reactor-year. The estimated probability of considerable fuel damage leading to the necessity of an evacuation of the population from the area prescribed by the relevant guides is specified to be less than 10^{-7} per reactor-year.

The safety of the nuclear power plant is ensured by a consistent implementation of the "defence-in-depth" principle based on the application of a system of barriers on the path of spreading ionizing radiation and radioactive substances into the environment, as well as of a system of engineered safeguards and of organizational provisions for the protection of these barriers. The consistent implementation of the "defence-in-depth" principle implies:

- Installation of successive physical barriers on the path of spreading the radioactive substances: fuel matrix, fuel element cladding, primary circuit boundary, containment;
- Taking into account all postulated initial events that can lead to a loss of efficiency of these barriers;
- Determination, for each postulated event, of design measures and actions of operating personnel required to keep the integrity of the barriers mentioned, and to mitigate the consequences of a damage of such barriers;
- Minimization of the probability of accidents resulting in an escape of radioactive substances; and
- Redundancy and diversity of safety systems, and physical separation of safety system trains.

The nuclear power plant considered is of the evolutionary type. The principal technical decisions have been supported by operating experience for about 1000 reactor-years with nuclear power plant of the WWER-440 and WWER-1000 type, including the Loviisa and Paks plants in Finland and Hungary which are known to be among the best performing nuclear power plants in the world measured by their load factors.

The new design features are envisaged to be verified experimentally at a large-scale test facility (1:27 volume and power scale). The design is developed in accordance with the latest Russian safety regulations for nuclear power plants [1,2] which meet modern world requirements. The design organizations involved are: EDO "Gidropress", Russian National Research Centre "Kurchatov

Institute" and SPAEP, the well-known designers of WWER nuclear power plants. QA requirements of the IAEA and the international standard ISO 9000 are taken into account in the design.

In the plant safety concept, the world's modern trends in nuclear power plant safety improvements are considered in order to meet, for as long a period as possible, the current and future requirements for nuclear power plant safety which are constantly becoming more strict. The design passed the international examination at the nuclear power plant design competition in St. Petersburg in 1993. On the 7th of June 1996 Gosatomnadzor of Russia issued the permit for constructing a forerunner of the power unit with VVER-640 at the Kola-2 NPP. On the 25th of June 1996 Gosatomnadzor of Russia (Russian Nuclear Regulatory Commission) issued permit for constructing a forerunner of the power unit with VVER-640, being a part of the Scientific-and-Industrial Center of Nuclear Power in Sosnovy Bor.

Apart from the review in the appropriate Russian Supervisory Authorities to confirm that the accepted solutions correspond to the world safety criteria and requirements, the design was subjected to a comprehensive examination by the leading foreign experts from Germany (GRS) and USA (Brookhaven, Argonne and Battelle Laboratories, Raytheon Engineers and Costructors, Bechtel Power Corporation).

5.5.2 Description of the nuclear systems

5.5.2.1 Primary circuit and its main characteristics

The design of the primary circuit uses a 4-loop configuration with horizontal steam generators. The flow diagram and a drawing of the reactor plant are shown in Figures 5.5-1 and 5.5-2. The list of the components shown in Figure 5.5-1 is given in the Table 5.5-I.

5.5.2.2 Reactor core and fuel design

Reactor core and fuel design is the same as applied in the advanced WWER-1000 designs. The core comprises 163 fuel assemblies. In the reactor scram system 121 control rods are used. A step-by-step electromagnetic (magnetic jack) drive with position indicator is used to move the control rods. The control rod assemblies comprise 18 absorber rods suspended on a brace. The absorber rods are made of stainless steel tubes filled with an absorber material which varies along the length, boron carbide is used for the upper part and dysprosium titanate for the lower. The rod cladding is sealed with end caps. The control rod drives are installed on the reactor top head. Effective operation time between refuellings is 293 full power days. Average burnup of the fuel unloaded is 40 MWdays/kgU. The number of fresh assemblies loaded during annual refuelling is 36.

5.5.2.3 Fuel handling and transfer systems

The fuel handling and transfer systems are intended for loading of fresh fuel assemblies and replacing spent fuel assemblies, and shutdown absorber rods. The following compartments with necessary equipmentare foreseen: the reactor concrete well; the fuel pool; and the transfer compartment.

The fuel pool is located in the vicinity of the reactor concrete well; they are connected by a transfer channel designed for transportation of one fuel assembly at a time. The fuel pool is provided with storage structures for spent fuel assemblies; these structures consist of separate sections designed for storing of fuel assemblies and sealed containers for failed fuel assemblies.

The transfer compartment is provided with an universal nest for location of a plant-internal transportation unit for fresh fuel assemblies and for a transport cask for spent and decayed fuel assemblies. The transfer compartment is connected to the fuel pool by the transfer channel which the

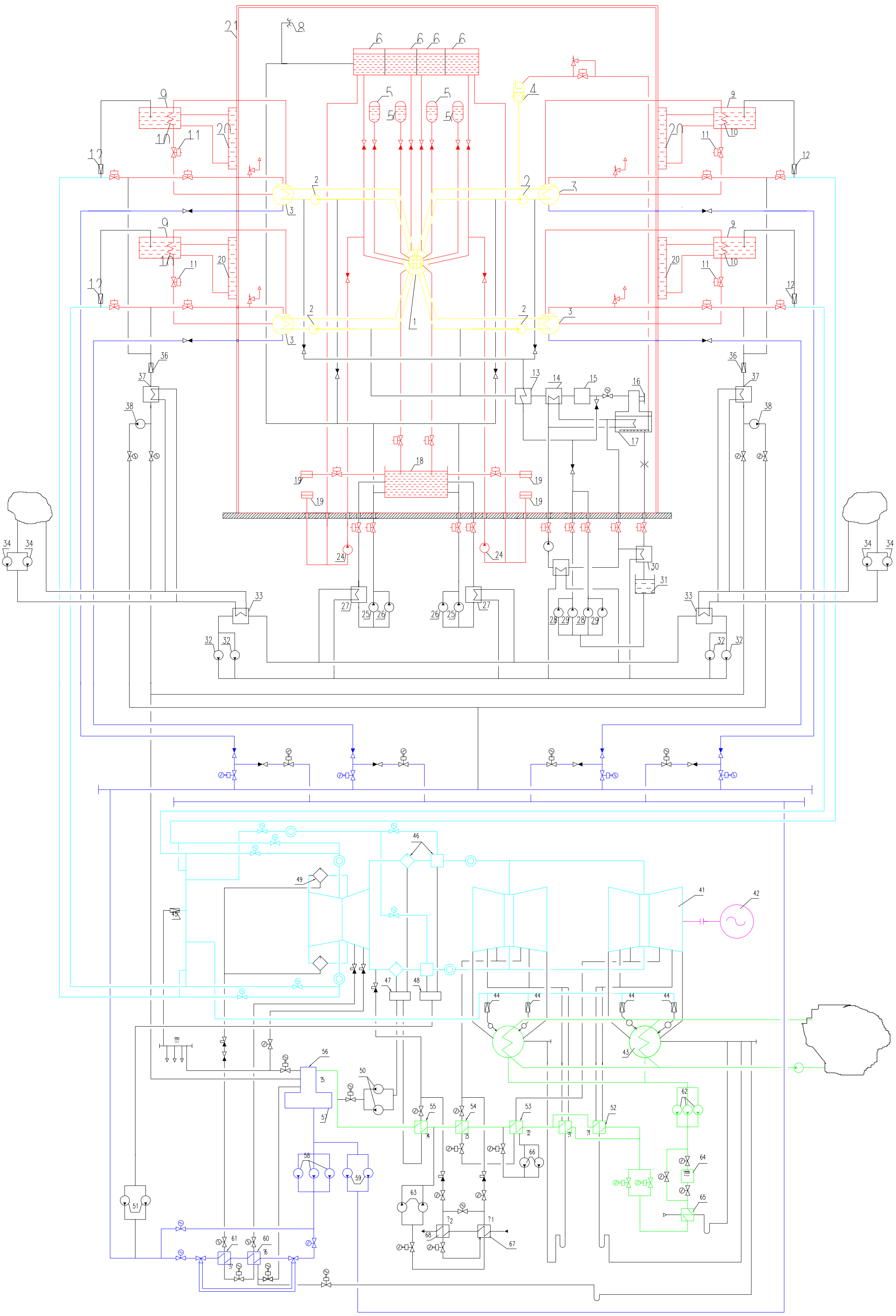

FIG. 5.5-1. Flow diagram of the NPP system.

(See Table 5.5-I for the correspondence of the numbering to the equipment.)

TABLE 5.5-I. LIST OF THE REACTOR BUILDING SYSTEM EQUIPMENT

1.	Reactor	$N_{\text{т}}$=1800 MW
2.	Reactor coolant pump	Q=12800 m^3/h
3.	Steam generator	$N_{\text{т}}$=450 MW
4.	Pressurizer	V=79 m^3
5.	ECCS accumulator tanks	P=4.0 MPa V=75 m^3
6.	ECCS tank	V=500 m^3
8.	Spray header	
9.	Emergency heat removal tank	V=500 m^3
10.	PHRS heat exchanger	F=210 m^2
11.	PHRS starting valve	D_{nom} 175
12.	QPRP-A	G=540 t/h
13.	Regenerative heat exchanger	F=107 m^2
14.	After-cooler	F=43 m^2
15.	Special water treatment plant	P_p=15.9 MPa
16.	Rupture membrane	P_p=1.2 MPa
17.	Bubbler-deaerator	$V_{\text{полн}}$=36 m3
18.	Fuel pool	V=500 m^3
19.	Screen filter	S=0.125 m^2
20.	Tight space cooler	F=5 m^2
21.	Containment	V=57000 m^3
24.	Primary circuit emergency injection pump	Q=20 m^3/h H=1650 mH_2O
25.	Fuel cooling system pump	Q=800 m^3/h H=70 mH_2O
26.	Fuel cooling system pump	Q=300 m^3/h H=80 mH_2O
27.	Fuel cooling system heat exchanger	F=1000 m^2
28.	Primary circuit makeup pump	Q=20 m^3/h H=1600 mH_2O
29.	Primary circuit makeup pump	Q=6.0 m^3/h H=1600 mH_2O
30.	Coolant removal heat exchanger	F=23 m^2
31.	Damper tank	V=10 m^3
32.	Pump of intermediate cooling circuit of reactor building consumers	Q=800 m^3/h H=70 mH_2O
33.	Heat exchanger of intermediate cooling circuit	F=1000 m^2
34.	Cooling water pump	Q=800 m^3/h H=70 mH_2O
36.	Cooling down pressure reducing plant	
37.	Process condenser	F=442 m^2
38.	Cooling down pump	Q=800 m^3/h H=70 mH_2O
41.	Steam turbine	70 kgs/cm^2 284.5 ^oC
42.	Generator	$N_{\text{э}}$=645 MW
43.	Condenser	F=44000 m^2
44.	QPRP for steam discharge to condenser	550 t/h
45.	Quick-acting pressure reducing plant (QPRP) of auxiliary steam supply	180 t/h
46.	Separator-superheater (SSH)	F=600 m^2
47.	SSH separated moisture collector	
48.	Condensate collector	
49.	Primary separator	
50.	SSH separated moisture pump	Q=500 m^3/h H=200 mH_2O
51.	SSH condensate pump	Q=400 m^3/h H=400 mH_2O
52.	Low pressure heater No1	F=990 m^2
53.	Low pressure heater No2	F=1145 m^2
54.	Low pressure heater No3	F=1766 m^2
55.	Low pressure heater No4	F=2091 m^2
56.	Deaeration column	Pp=1.32 MPa
57.	Deaerator tank	V=210 m^3
58.	Feed water pump	Q=1470 m^3/h H=900 mH^2O
59.	Auxiliary feed water pump	Q=250 m^3/h H=830 mH_2O
60.	High pressure heater No6	F=2560 m^2
61.	High pressure heater No7	F=2560 m^2
62.	Main condensate pump	Q=1000 m^3/h H=315 mH_2O
63.	Condensate pump of unit for heat supply	Q=500 m^3/h H=220 mH_2O
64.	Drain expansion tank	G=100%
65.	Gland seal steam heater	F=220 m^2
66.	Drain pump of l.p. heater No 2	Q=270 m^3/h H=210 mH_2O
67.	Main network heater	F=1000 m^2
68.	Peak-load network heater	F=1000 m^2

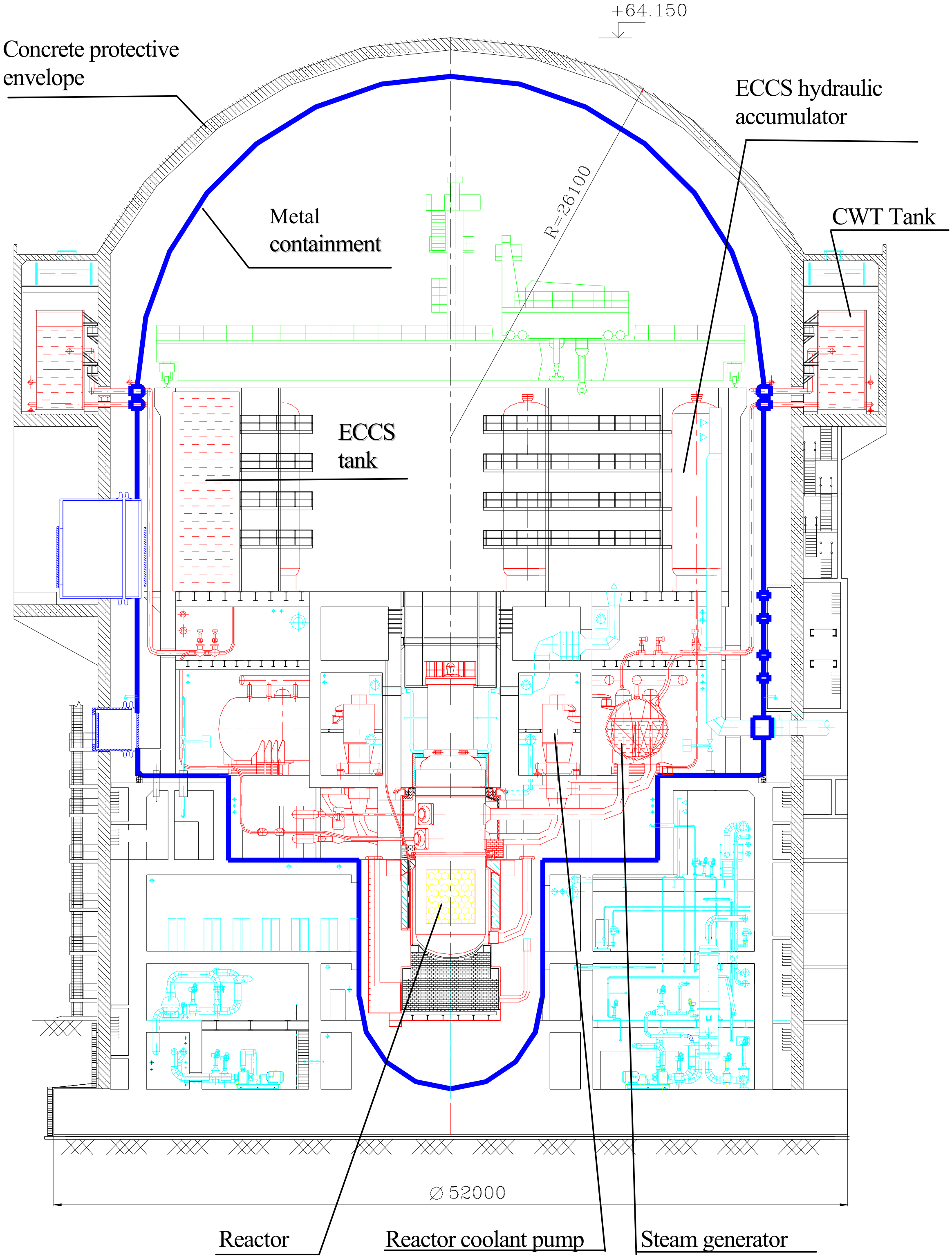

FIG. 5.5-2. Reactor building.

loading of fresh fuel assemblies takes place. The fresh fuel assemblies are taken out of the plant transportation unit by the refuelling machine and installed in the core in accordance with the core refuelling chart. After unloading from the core, the spent fuel assembly is placed in a container with a fuel assembly defect monitoring system, and then transported to the storage structure or the sealed container for spent and decayed fuel assemblies in the fuel pool depending on the results of the defect inspection.

Manipulations for control rod assembly replacements are similar to those for fuel handling; control rod clusters and burnable poison rods extracted from the reactor core are installed in empty fuel assemblies or in storage structures in the fuel pool for storage. The refuelling machine handles only one fuel assembly, one control rod cluster, or one fuel assembly with the control rod cluster inside it, at a time. Heat removal from the reloaded fuel assemblies is accomplished by the fuel pool cooling system.

5.5.2.4 Primary components

Reactor pressure vessel

The reactor vessel of the earthquake-proof WWER-1000/V-392 design is adopted for the V-407 reactor. The vessel design lifetime is 60 years; for the rest of the reactor equipment, except equipment that is exchangeable during the service life (in accordance with its prescribed lifetime), the design lifetime is 50 years.

The reactor assembly consists of the following parts: reactor vessel with main closure head and associated details; upper unit with 121 control rod drive mechanisms; reactor internals (core support structures, outer core structure, upper protective tube structure); and the reactor core. The reactor vessel consists of a flange, upper and lower nozzle zone shells, support shell, cylindrical shell and elliptic bottom, welded together by circumferential seams. The vessel has two rows of nozzles of 620 mm inner diameter; four nozzles in each row. In addition, there are four nozzles of 170 mm inner diameter for the pipeline connections to the ECCS hydroaccumulators in the upper row and four nozzles of 125 mm inner diameter for the pipeline connections to the ECCS tanks in the lower row; the upper row also includes an instrumentation nozzle.

The inner surfaces of the vessel and the nozzles are plated with a corrosion-resistant layer. A separation ring is welded to the vessel inside between the upper and lower nozzle rows for separation between the inlet and outlet coolant flows, and a support rib is made on the vessel support shell for vessel attachment on the support structure. The reactor vessel is installed in the concrete well on the support framework; the support rib of the vessel is installed and fixed on the support framework ring.

Reactor internals

Structures providing support and positioning of the core and the control rods are installed inside the vessel; the internal structures including the core support structures, outer core structure, and upper protective tube structure also serve as coolant flow guides for the core heat removal.

Steam generators

The steam generator of advanced WWER-1000 designs is adopted for WWER-640 plant.

Pressurizer

The pressurizer is the same as for the WWER-1000 design. The pressurizer with primary pressurization system is designed for maintaining primary system pressure within acceptable limits in all reactor plant design conditions. The pressurizer proper is a vertical vessel mounted on a cylindrical support structure. There are nozzles for spray pipelines and safety valve steam discharge line on the upper head, and a nozzle for connection between the pressurizer and the hot leg is provided in the vessel bottom.

A spray device and a tubular electric heater are located inside the pressurizeer. The spray device is intended for water spraying in the steam volume and condensation of steam; it is made as a discharging header fastened to the top of the vessel. The inside surface of the pressurizer is plated with corrosion resistant stainless steel. All pressurizer internals are made of austenitic stainless steel.

The primary system overpressure protection is designed to avoid inadmissibly high coolant pressure in accident situations. The system can also be used for forced depressurization of the primary system. It consists of two safety valves, a relief valve and pipelines connecting them with the pressurizer; and a bubbler-condenser and pipelines connecting it with the safety and relief valves.

Reactor coolant pumps

The reactor coolant pump (RCP) is of the centrifugal type in a spherical case. Lubrication and cooling of the RCP are performed with water. A non-combustible lubricant is used in the electrical motor. The pump flow path is arranged so that the loop seals in RCS are avoided.

Main coolant lines

The main coolant lines consist of four loops for coolant circulation. The inner diameter of the main pipelines is 620 mm. The loops do not have loop seals and consist of straight tubes and steep bends on sections of connections of steam generators and main coolant pumps. The primary equipment layout and the passive heat removal system provide core residual heat removal via the steam generators to the demineralized water storage tank and further to the atmosphere by natural circulation.

5.5.2.5 Reactor auxiliary systems

Chemical and volume control system

The primary circuit makeup water and boron concentrate storage system (KBC) is intended for receiving and storing the primary circuit makeup water (0 - 8 g H_3BO_3/kg H_2O) and boron concentrate (40 g H_3BO_3/kg H_2O). This system supplyies the water and boron concentrate to the primary circuit makeup-letdown system (KBA). The system active components are ensuring with electric power supply of the group 3. Exemption is made for the isolation valves installed on pipelines of drainage from the boron concentrate tanks to the KBA system - they are ensured with the electric power supply of the group 2 from diesel-generators. The system consists of the makeup water reserve tanks with the boron concentration up to 8 g H_3BO_3/kg H_2O, the boron concentrate reserve tanks with concentration of 40 g H_3BO_3/kg H_2O, makeup pumps, pipelines and valves.

5.5.2.6 Operating characteristics

The nuclear power plant unit is designed taking into account the grid needs imposing a high capability to follow grid loads in conditions with rather rapid and deep plant load changes.

5.5.3 Description of turbine generator plant systems

5.5.3.1 Turbine generator plant

The selected materials, reliable technical decisions on the equipment and process systems allow to carry out the equipment overhaul once in 6 years and prolongate the main equipment service life up to 50 years. The turbine plant availability factor is ensured more than 0.96. Turbine set is a single-shaft unit with one high and intermediate pressure cylinder (HIPC) and two double-flow low pressure cylinders (LPC). Low pressure cylinders are unified with the LPC of turbine K-1000-60/3000 being applied at WWER-1000 plants.

Using in the design of progressive technical decisions for the turbine design, auxiliary equipment and turbine plant layout, side by side with using the blade of 1200 mm length, well-established during operation in turbines of thermal and nuclear power plants, which is limiting one for high-speed turbines in the present time, allows to create turbine set, competitive with the foreign reference of the medium-power NPP at the world market.

Steam distribution is of throttle type. Two units of stop-control valves are available. The steam is supplying to every valve unit by two pipelines of 600 mm diameter. The steam supply to the turbine from every valve unit is performed by pipeline of the same diameter into the HIPC lower part. The high and intermediate pressure cylinders are combined in a single body with flow turn.

The LPC rotor is one-piece forged, mass with working blades is 82 t. All turbine rotors are connected with each other and with the generator by rigid couplings. The shaft line fixing point (the thrust bearing) is located between the HIPC and LPC-1. Every shaft line rotor is supporting on two bearings. With the purpose of preventing the bearing inserts wear at the shaft line rotation with the low frequency, provisions are made for the turbine and generator rotor hydraulic handling due to high pressure oil supply by the special pump under the rotor support neck of every bearing.

5.5.3.2 Condensate and feedwater systems

There are four low pressure heaters (LPH), increased pressure deaerator (1.32 MPa abs.), operating at varied pressure under the transients, and two high pressure heaters (HPH). The LPH-1, 2 and 3 are horizontal, arranged between LPC exhaust branch pipes. The LPH-1 is performed in two cases with parallel connection by the main condensate. LPH-2 is single-body one, arranged under LPC-1. The LPH-3 is also single-body one and arranged under LPC-2. LPH-4 is of vertical design, it is located in the turbine building.

For the plant three motor-operated main feed water pumps of PEA-1470-78 are accepted, which rated capacity is of 1470 m^3/h, head makes up 900 m of water, all three pumps are operating. Auxiliary feed water pumps of PEA-250-75-2 in amount of two pieces are participating in the power unit start-up and shutdown and are using at the scheduled cooldown condition.

As high pressure heaters HPH-6 and HPH-7, the heaters of HPH-C type are used. These were developed for turbine sets of power unit with VVER-1000 reactor plant. The heaters are of chamber type with the water chamber lower arrangement, vertical, equistrength by the steam, as well as by the water side. The U-shaped tubes of the stainless steel are expanded and welded in the tube plate. The HPH-C are installed in one line.

5.5.3.3 Auxiliary systems

The turbine is equipped with electrohydraulic automatic control system. The system includes: steam distribution elements; hydraulic part of the control system with the turbine acceleration preventive system and actuators; oil supply system for the control system hydraulic part; the control system electric part.

To keep the live steam pressure under the power unit startup and emergency operating conditions provisions are made for quick-acting pressure reducing plant to condenser (BRU-K) with capacity of up to 60% from the rated capacity of the reactor plant steam generators, which are including into the scope of control system hydraulic part.

5.5.4 Instrumentation and control systems

5.5.4.1 Design concept, including control room

The main control room is the main point point for monitoring the main processes of the nuclear power plant unit and for plant staff controlling it, in accordance with assigned operation tasks. The instrumentation and control design concept provides work positions for operators of the reactor, the turbine and auxiliary equipment, and for a shift supervisor responsible for coordinating the plant unit activities during all operation conditions.

The work positions for control of special engineering means ensuring plant safety, including auxiliary power supply systems, are separated from the normal work positions. A reserve control room for the plant unit is provided from which it is possible to ensure a reliable reactor shutdown to cold and subcritical state and maintain it in this state as long as necessary, including means for initiation of safety systems and devices for receiving information on reactor conditions. Plant process control systems fulfill the automatic control of the main plant parameters (neutron flux, primary and secondary pressure, pressurizer and SG level, etc).

5.5.4.2 Reactor protection and other safety systems

The monitoring and control system provides an automated diagnosis of the state and the operating conditions of the NPP. Monitoring and presentation of information is carried out on the reactor coolant system, on the containment, on all the systems important for safety under all operating conditions of the NPP. Remote control of these systems is possible. The operating personnel monitors the NPP systems as well as the parameters defining the NPP safe status in accordance with the service manuals from the main control room (MCR). Engineered features of the on-line diagnosis system are provided to give a possibility for an operator to form a correct estimate of the plant state, and to take necessary measures during and after an accident.

The automatic safety systems comprise:

- reactor scram system
- primary overpressure protection system
- emergency core cooling system
- system of passive heat removal from the reactor plant
- system of passive heat removal from the containment
- system of quick-acting isolation valves in steamlines
- secondary overpressure protection system
- diesel-generators
- system of reliable direct current power supply.

Reactor scram system

The power control and protection system is designed for generating and executing the commands for limiting or decreasing reactor power, or for reactor shutdown when any accidents occur as a result of reactor plant failures or of operator errors. The following types of preventive and emergency commands are foreseen:

- sequential movement of control rod groups downwards with nominal speed up to the disappearance of the emergency signal (first type of preventive protection);
- prohibition of control rod upwards movement (second type of preventive protection);
- drop of one control rod group (accelerated preventive protection);
- drop of all control rods (reactor scram).

To eliminate the consequences of severe (beyond-design) accidents with the control rods assumed failed, injection of a boric acid solution with a boron concentration of 16 g/l into the hot legs is provided from ECCS tanks by two independent systems. The pumps are started up in 15 s after the voltage is applied.

In case of the main control room (MCR) failure the reserve control room (RCR) is used to provide reactor shutdown, monitoring of subcriticality, etc. The possibility of control of the systems important for safety is retained from the RCR. Autonomous habitability under conditions of unavailability of regular ventilation systems is provided for the reserve control room for design events including safe shutdown earthquake (SSE) and, connected with it, fire and other site damage.

5.5.5 Electrical systems

5.5.5.1 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 aspects and the time interval of loss of electric power (fraction of second; time to be specified by safety conditions for various groups of equipment without increased requirements).

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 the main and the reserve grid connections, is carried out in a time not exceeding 15 s from the moment of generation of a command for start up.

The DC electric power supply of the reactor control and protection system is ensured by accumulator batteries (in each channel) designed for a discharge over 24 hours. Electric power from the accumulator batteries during a station blackout is provided for both the MCR and the RCR in full measure.

5.5.5.2 Safety-related systems

Two physically separated diesel generators provide power to the safety related systems for 2 days using its own inventory of fuel, and for an unlimited time when fuel is delivered from the outside. The system of reliable direct current power supply comprising storage batteries, provides the power supply to electromagnetic circuits for the operation of automatic safety systems as well as for recording of necessary parameters during 24 hours.

5.5.6 Safety concept

5.5.6.1 Safety requirements and design philosophy

The principal features characterizing the safety philosophy accepted in the design are as follows:

- Considerable decrease of specific fuel power (it is 1.25 times less than in the Loviisa nuclear power plant reactor and 1.5 times less than in a standard WWER-1000 reactor) due to an increased number of fuel assemblies;
- The fluence to the reactor vessel considered is one order of magnitude less over 60 years than that to the vessel of the standard WWER-1000 reactor over 40 years;
- The possibility of providing subcriticality with solid control rods at any moment of the life-time for a coolant temperature decrease down to 100 °C and assuming complete replacement of the boric acid in the primary circuit with pure condensate;
- Retaining the large ratio of the primary and secondary coolant volumes to the reactor power typical for WWER-440 reactors (1.5-2 times more in comparison with the standard WWER-1000 reactor and Western PWRs;
- Simplification of the operating and layout features for safety systems and all other systems of the plant (in comparison with a WWER-440 the number of pumps and compressors is reduced 4 times, the number of shut-off valves 3 times; high and low pressure tanks 2 times, and the number of sealed process penetrations is reduced by 4 times);
- Application of horizontal steam generators with large water inventories and better conditions for natural circulation in the primary circuit in comparison with vertical steam generators;

- Application of an emergency core cooling system based on the principle of passive operation that provides for the possibility of long-term residual heat removal after LOCA accidents taking also into account a possible station blackout;
- Application of passive systems for residual heat removal from the reactor plant in case of a station blackout (transient);
- Application of passive systems of residual heat removal from the containment;
- Provision of a large water inventory inside the containment (about 2000 m^3) required to form the emergency heat removal pool, the water level of which rises above the hot legs after flooding;
- Application of an inner, sealed steel shell, enclosed by an outer concrete protective shell, and both together constituting the containment system;
- Application of diagnosis systems for equipment and systems important to safety for on-line diagnosis during operation and for periodic inspections after shutdown;
- Application of an automatic control system of improved reliability, with self-diagnosis, and an expert system for giving advice to the operator;
- Redundancy, diversity, physical separation of safety systems as part of defence in depth.

Deterministic design basis

The design applies a deterministic approach presupposing that in the design the safety under normal operating conditions and at any initiating event included into the design, taking into account a single-failure principle, is provided for. Besides, the measures on management of beyond design basis accidents and mitigation of their sequences are also provided for in the design.

Risk reduction

The design is worked out on the basis of using the passive safety systems and technical decisions oriented to prevent the development of initiating events into an accident.

In order to prevent the radiation uncontrolled release into environment at power unit operation, the following technical decisions are realized in the design:

- Separate outer containment, which ensures external effect protection and underpressure in intercontainment space;
- Closed intermediate circuit with a pressure being below the pressure in external cooling water system, for cooling the equipment with radioactive media;
- Purification system intended for elimination of radioactive products in gaseous (ventilation air and process vents) and liquid media under all NPP operating conditions, inclusive the accidents;

External and internal hazards

The design is performed taking into account two levels of seismicity: an operating basis earthquake (OBE) of magnitude 7 on the MSK-64 scale and a safe shutdown earthquake (SSE) of magnitude 8 on the MSK-64 scale.

During operating basis earthquake, normal operation of the reactor plant is ensured. During the safe shutdown earthquake, reactor and plant shutdown, cooling and fuel discharge are ensured. All civil structures, process and electrotechnical equipment, pipelines, instrumentation, and so on, are divided into three seismic categories depending upon the degree of responsibility for safety ensurance during seismic events and on the serviceability after the earthquake. Components and systems of category 1 (the highest) can fulfill their safety functions during and after an earthquake of SSE intensity. The design considers the possibility of using special seismic isolators located under the base plate.

The external wind load for the first category buildings and constructions is assumed to amount to 0.9 kPa, corresponding to a hurricane wind speed of 38 m/s. Effects of a tornado are taken into account in the design for the first category of buildings and structures (maximum horizontal speed of rotation of the tornado wall 60 m/s, speed of the tornado 15 m/s, tornado radius 50 m, wind front pressure 3.5 kPa, pressure differential between the center and the periphery 4.4 kPa, speed of missiles carried by tornado 20 m/s).

As for the external industrial hazards and airplane crash, the design is based on front pressure of the assumed explosion shock wave 30 kPa and impact of a plane with 5.7 t mass at a speed of 100 m/s.

5.5.6.2 Safety systems and features (active, passive, and inherent)

Safety systems configuration

The residual heat under non-LOCA is removed by normal operation systems, having standby electric power supply from the unit diesel-generator station in case of loss of off-site and on-site power. In case of the unit diesel-generators failure, the passive heat removal system from steam generators is putting into operation. This system ensures the reactor plant cooldown and long-term heat removal owing to steam condensation in heat exchangers, submerged under the water level in the emergency heat removal tank. The condensate is returning to the steam generators. In case of a LOCA, the boric acid solution is supplying to the reactor from accumulator tanks with initial pressure of 4.0 MPa. If functioning of pumping devices, ensuring the primary circuit injection, is not providing, there is following the core filling up from the boric acid solution emergency reserve tanks, located inside the containment. The coolant, releasing through the break, is collecting on the containment floor in the specially organized enclosure (emergency heat removal pool). For guaranteed discharge of ECCS tanks through the core into the emergency pool, at pressure decreasing up to the setting of the primary circuit pressure - 0.6 MPa, automatic opening of the untightening valves occurs, connecting the reactor with the refuelling pool. As the primary circuit, emergency accumulators and tanks are discharging and the emergency pool level is increasing over the level of the reactor inlet branch pipes. As a level is forming in the emergency pool, as opening of the valves, connecting the refuelling pool with the emergency pool occurs. In the course of the accident development the possibility of heat removal via the passive heat removal system through steam generators is taking into account only during the period, when the steam generator headers are filled up with water. The steam cooldown phase is considering as conservative margin.

After the discharge of primary circuit, ECCS accumulator tanks and ECCS tanks the natural circulation circuit is forming: the core-the reactor upper mixing chamber-the untightening valve above the core-the refuelling pool- the untightening valve under the core-the reactor lower mixing chamber-the core. Heat removal from the refuelling pool occurs owing to evaporation of the coolant part with the following steam condensation on the walls and within the scope of the steel inner containment and condensate returning into the refuelling pool, as well as into the emergency pool. The refuelling pool makeup is carried out from the emergency pool via the valve, connecting the refuelling pool with the emergency pool. To ensure necessary tightness of the emergency heat removal pool all penetrations through the containment are arranging at elevations above the maximum level in the pool. The containment heat removal is carrying out through the wall to the water of the containment heat removal system. The steam, available in the containment, is condensing on the heat conduction inner surfaces of the containment and most intensively in the places of the guide-vane cooling channels location. As a result of these processes the coolant is heating in the guide-vane cooling channels, that causes inducing the lift and the coolant movement at the natural circulation circuit grade section, cooling the containment from the outside. The cold coolant is supplying by the descent section from the emergency heat removal tank to the guide-vane cooling channels. In the course of the system operation water is gradual heating and partial boiling in the emergency heat removal tank. Water inventory in the tank is designed for heat removal during no less, than 24 hours, with consideration of the single failure criteria.

In accordance with the safety concept for the post accident measures it is provided, that residual heat will be removed after 24 hours by using the cooling systems of safety-important normal operation consumers, having constant water source and reliable electric power supply. If it is impossible to recover functioning of the systems, ensuring normal heat removal from the reactor plant and refuelling pool during up to 24 hours (for instance, at the beyond design basis accidents), the personnel ensures connection of trains, not participating at emergency cooldown, or makeup of emergency heat removal tanks by using the power unit stationary or movable systems, for example, fire fighting systems.

Safety systems ensures a very low frequency of severe core damage (significantly less than of 10^{-6} per reactor-year). Nevertheless, taking into account modern tendencies of extension of defence in depth methods, there is developing assurance of the corium confinement within the reactor pressure vessel and beyond its limits. Nonreaching of the explosion hazard hydrogen concentration in the gas-steam mixture is ensuring by its controlled burning under the containment.

Passive heat removal system via steam generators

Passive heat removal system via steam generators (PHRS) is intended to remove decay heat from the reactor under complete loss of power sources with intact primary and secondary circuits, and to depressurize the RCS in case a small break LOCA. The PHRS removes the heat to the heat exchangers immersed to the emergency heat removal tanks, which are installed outside the containment. The PHRS ensures heat removal from the reactor plant during at least 24 hours (total time of autonomous system operation is 72 hours). This system consists of four channels, each channel having 50% performance. Each channel has emergency reserve water tank, heat exchanger, valves, and pipelines.

Each system's channel is connected to steam and water volumes of the relevant SG. The valves of 175 mm 50 mm diameter are installed on the condensate pipeline. Both valves have passive drive and ensure system automatic switch-on. Besides, isolation maintenance valves are installed on steam supply and condensate removal pipelines. The valve 50 mm opens under station blackout mode to remove decay heat from the reactor plant. The valve 175 mm opens in case of a SBLOCA for fast depressurization of the primary circuit (and ECCS actuation).

ECCS accumulator system

ECCS accumulator system is intended for the cooling water supply at the core inlet and for keeping the coolant volume in the primary circuit under conditions of leakages with the primary circuit pressure less than 4.0 MPa. The system consists of four independent trains with redundancy of 4x50%. Each train includes accumulator tank with boron solution reserve under the pressure, being created by the nitrogen cushion, valves and pipelines. The accumulators are connected to the reactor downcomer. The check valves are installed on the connecting pipelines to cut off the accumulators from the reactor when the primary pressure exceeds 4.0 MPa. The system is operating in accordance with the passive principle, and no signal is required to activate the system.

ECCS tank system

ECCS tanks (tanks of atmospheric type) are intended for the reactor core flooding with the aim of long-term residual heat removal and they are operating at the primary circuit pressure below 0.29 Mpa. The system consists of four independent trains with redundancy of 4x50%. Each train includes the tank filled with boron solution, valves and pipelines. The ECCS tank is connected to the reactor downcomer by the pipeline with two check valves. The system is operating under the hydrostatic force effect.

Primary circuit untightening system

The primary circuit untightening system is designed to ensure reliable water flow from atmospheric ECCS tanks and natural circulation of coolant along the circuit "reactor- emergency pool". This system operates at LOCAs when the primary circuit pressure decreases to 0,6 MPa. The system consists of two independent trains with redundancy of 2x100%, and each train comprises two pipelines and valves, connecting the hot and cold legs of the primary circuit to the emergency pool located in the lower part of the containment. This will ensure the long-term residual heat removal after the ECCS accumulator tanks and atmospheric tanks are emptied. The system train includes untightening valve units, pipeline and valves for connecting the refuelling pool with the emergency pool, repair valves, screen filter.

Emergency gas removal system

The emergency gas removal system is intended for removal of non-condensing gases from the pressurizer upper part, the steam generator header upper part; the reactor top head to prevent breakdown of the natural circulation. The system includes isolation valves and pipelines connected to the bubbler-deaerator of the makeup-letdown system. Two isolation valves are installed in parallel on each pipeline. The valves are energizing from different trains of the emergency electric power supply system.

Primary circuit emergency injection system

The system is designed to ensure the core subcriticality in case of failure of the control rods to scram the reactor (ATWS). This system also contributes to core cooling in some scenarios providing the diverse mean for water supply to the primary circuit.

Localizing safety systems

The protective tight steel shell is intended for confinement of radioactive releases at the design basis accidents and beyond design basis accidents with the primary circuit untightening, including break of the steam generator primary header. The maximum pressure under the shell is specified outgoing from the conditions of tightness keeping during the overpressure period. The design leak makes up 0.2% from the air volume per day at the design pressure. The shell is performed as a cylinder with the hemisphere dome. The cylindrical part diameter is 41.0 m, height is 30 m, thickness of the cylindrical shell wall is 36 mm.

Selection of the shell design characteristics is carried out corresponding to the loads and their combinations, occurring at the normal operation conditions, pneumatic tests, emergency conditions and accidents, and also loads, induced by the external effects (earthquakes, air crash, explosion near the reactor building). The combination of the design basis accidents and design basis earthquake is taken into account in the design of the protective steel shell.

The protective tight shell is also important element of the containment heat removal system. During accidents with the primary and secondary circuits untightening the thermal power is removing from the containment by the protective shell wall owing to steam condensation from the gas-steam mixture. From the external surface of the shell cylindrical part the heat is removing by using the coolers of the passive heat removal system. The coolers are arranged in the cylindrical part area.

Containment passive heat removal system

The containment passive heat removal system is intended for emergency heat removal from the containment to the environment in case of a LOCA. The system does not require the electric power supply sources for its functioning. The system consists of four independent trains with the redundancy of 4x33%. The system train consists of a range of guide-vane trains, located on the steel shell, filled up with water and connected to the emergency heat removal tank.

Emergency hydrogen suppression system

The emergency hydrogen suppression system is intended for preventing the explosion hazard concentrations and non-organized hydrogen burning in the NPP containment rooms under emergency conditions, including beyond design basis accidents (severe accidents) with the fuel melting. The system consists of a set of water-repellency treated catalysts, arranged within the tight space.

5.5.6.3 Severe accidents (beyond design basis accidents)

Severe accident mitigation strategy

In the WWER-640 design, the consecutive strategy has been developed to prevent the arising of accident and if failed – to make a possibility to manage severe accidents. In this case the main emphasis is made for provision of a leak-tight containment, since at considerable damage of the containment a great release of radioactive substances into the environment cannot be excluded.

Severe accident prevention and mitigation features

Based upon the results of deterministic and probabilistic analyses, the design substantiates and specifies the list of beyond design basis accidents in consultation with the national regulatory body. The design includes the engineering and organization measures to prevent the containment failure. Therewith, radiation effect on personnel, public and environment is limited.

In the design, the hypothetical core melt accidents with the probability less than 1.0E-7 per reactor-year are also considered. Such accidents are postulated as the design basis for the engineering measures as well as for the emergency operating procedures. The plant is equipped with the effective systems localizing the radioactive products released from the core during the postulated core melt sequences. These systems are as follows:

- Double protective containment with design untightness of internal shell not more than 0.2 % daily volume at maximum overpressure 0.4 MPa;
- ECCS complex capable to keep the corium inside the reactor pressure vessel due to emergency pool formation and reactor pressure vessel bottom cooling;
- Core catcher, which prevents the corium from spreading outside the containment.

Additionally, the normal operating systems listed below participate in emergency release decreasing at conditions of power supply presence:

- Ventilation system, equipped with filters, for maintaining the underpressure within the intercontainment space;
- Fuel cooling system;
- System of fuel pool water purification.

The technical measures ensure that the probability of accident requiring the urgent protective measures for public (evacuation and migration) is well below 1.0E-7 per reactor-year. Postulated beyond design basis accidents at the power unit with reactor plant VVER-640 (estimated emergency release occurrence values up to 1.0 E-8 1/year) are not beyond the IAEA scale level “5” (accident followed by risks outside the site).

5.5.7 Plant layout

5.5.7.1 Buildings and structures, including plot plan

In order to minimize the site area and communications, power unit buildings are located practically adjacent to one another, and arranged around the (central) reactor building, since they are functionally tied to the primary circuit systems. The layout and arrangement of buildings are designed based on functions of the buildings, process and transportation communications, and a concept of splitting all the premises into two principal areas.

Reactor and turbine buildings containing power unit principal equipment are dominant in the building complex. They are located along the power unit axis, with a clearance between them needed for service and transportation communications. Adjacent to the reactor building side is the auxiliary building containing special water purification (SWP) system equipment. Electric and control buildings are located adjacent to the turbine building.

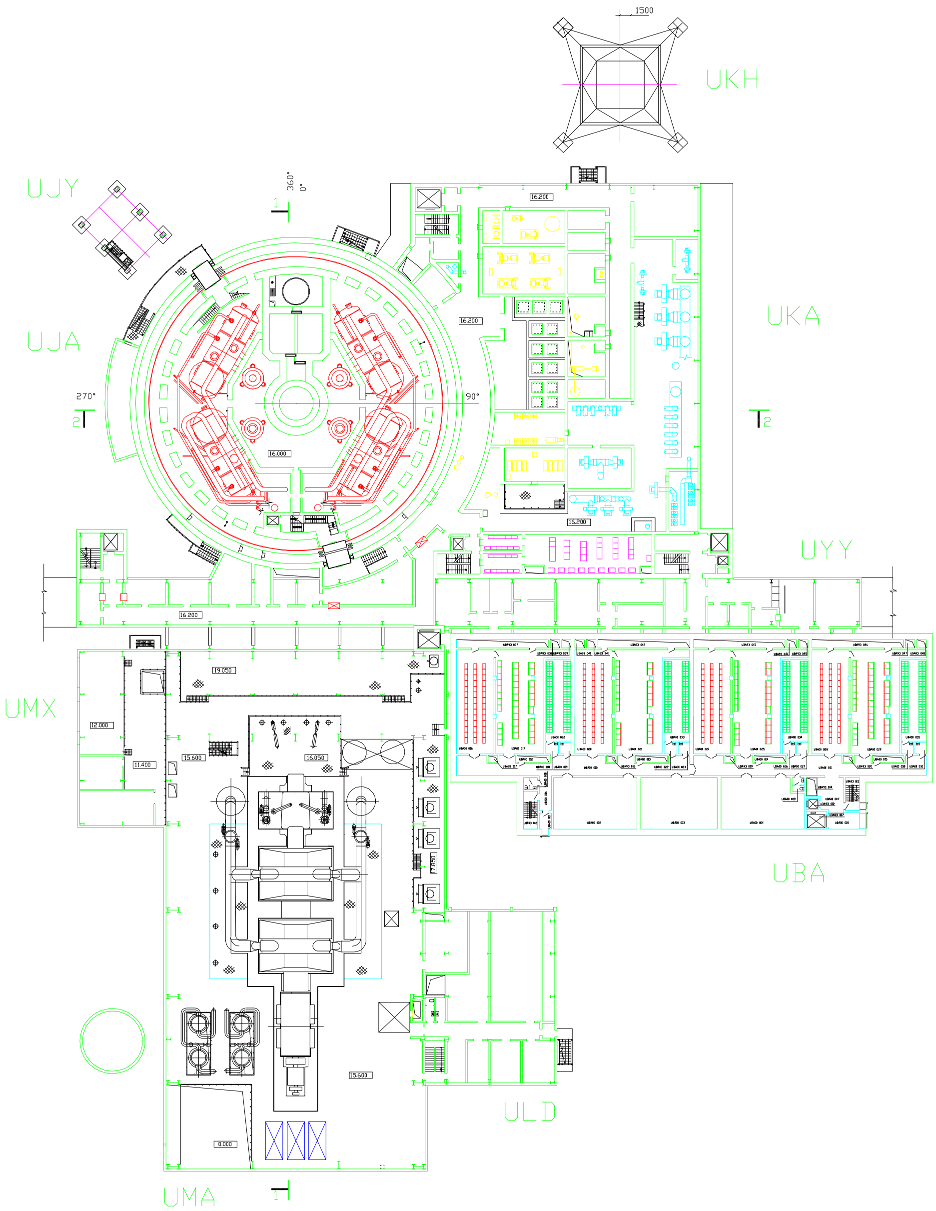

FIG. 5.5-3. Power unit layout.

UJA	Reactor Building	ULD	Condensate Purification Building
UKA	Auxiliary Building	UMX	Cooldown and Oil Facilities Building
UBA	Control Building	UJY	Transport Trestlework
UYY	Trestlework Building	UKH	Ventilation Stack with Air Ducts
UMA	Turbine Building		

The buildings and constructions are classified according to their safety importance in compliance with applicable Codes. Structures, nodes, equipment, and systems classified in Seismic Resistance Category I (the highest) must perform their safety functions during and after SSE, and retain their functionality during and after OBE. Structures, nodes, equipment, and systems classified in Seismic Resistance Category II must retain their functionality during and after earthquake of any intensity up to OBE. Seismic Resistance Category III includes structures design so as to stand OBE according to General Construction Codes (SNiP).

Seismic Resistance Category I covers:

- Normal operation systems and their components, whose failure due to seismic loads equivalent to SSE may result in radioactive release involving population exposure beyond the limits specified for the most severe design basis accident;
- Safety systems responsible for maintaining core sub-criticality, emergency cooling of the reactor, and fission products' isolation;
- Structures, buildings, equipment, and components thereof, whose physical damage caused by earthquake ranged up to SSE may result in their failure.

Seismic Resistance Category II covers structures, buildings, equipment, and components thereof (not included in Category I), whose abnormal operation may result in break in power production and/or radiation dose exceeding allowable annual value for NPP normal operation. Seismic Resistance Category III covers all the rest structures and edifices (not included in Seismic Resistance Categories I and II).

5.5.7.2 Reactor building

Reactor building falls into Category I in terms of seismic resistance and radiation/nuclear safety importance, all the related requirements being applicable. The building consists of the plinth located outside the containment, whereupon the internal steel shell is mounted, which, together with plinth top, forms the low-leakage accident localization space. The internal protective shell together with the plinth is in turn enveloped by an external concrete enclosure, annular space being provided between the both structures. The external concrete enclosure together with the plinth rests on a common foundation.

Because of extremely high requirements to reactor building structures, the project has provisions for their instrumental monitoring on permanent basis in order for the personnel to constantly be aware of their conditions and reliability status. Also the project provides permanent monitoring of the containment (for integral leakage), spent fuel pit (for leakage), and the building (for subsidence).

The walls and floors are reinforced with single bars, twisted-steel fabric, and welded reinforcing mats. Ready made form is used for concreting walls and floors, while slipform is used for concreting the secondary containment. The wall lining can be used as formwork if properly fixed to stand inter alia loads from freshly-poured concrete and other concrete-related loads.

5.5.7.3 Containment

The reactor building is arranged as a spatial system consisting of external protective enclosure (secondary containment) and located therein plinth-supported part with internal steel protective shell (primary containment). The plinth-supported part may be split into structures located within and without the primary containment. The previous together with the steel protective shell itself form a low-leakage space. The external protective enclosure is basically a cylinder 46.2 m in diameter topped with a truncated spherical dome. External enclosure top elevation is 64.150. Section rooms containing PHRS tanks are located along the secondary containment external perimeter at elevation 39.500. Below (elevation 20.350) a 2-floor gallery is provided, which partly envelops the secondary containment. It is designed to facilitate system cabling penetrations into the low-leakage space of the reactor building. The plinth together with the primary containment and steel structures installed therein is covered with external protection enclosure (the secondary containment). There are no direct mechanical ties between the plinth and related structures, which prevents transmission of emergency external loads (such as aircraft crash or explosion wave) to the reactor building internal structures.

Reactor cavity is located in the centre of the reactor building between elevations –4.500 and 28.000. Adjacent to the reactor cavity is a fuel bay, its bottom elevation being 4.500. The fuel bay includes a space for fuel transportation section. The latter's bottom has an airlock through which fuel assemblies are delivered to and removed from the bay. Service hall is located at elevation 28.000. This floor consists of monolithic concrete area and steel structures. It supports the principal process equipment, like high pressure ECCS tanks, low pressure ECCS tanks, and reloading machine.

Two airlocks are located at elevation 16.000 adjacent to separated stairwells provide entries into the low-leakage space. One airlock provides the access for maintenance personnel during reactor outage. This airlock is tied with a separated stairwell and a passenger elevator, connecting all low-leakage space elevations and providing an exit. The second airlock provides an emergency path of escape in case of possible incidents involved in maintenance procedures during reactor outage (such as flames or fumigation).

The external enclosure provides protection of the internal steel enclosure together with primary circuit equipment contained therein. Basically it is monolithic concrete cylinder 44.6 m in diameter, with semi-spherical monolithic concrete cupola. There is a clearance between both shells 1800 mm in width (above elevation 16.000). Annular cantilever is envisaged at elevation 39.500 to support tanks of passive heat removal system (PHRS) of 2000 m^3 volume, and fire-fighting water inventory tanks. The external enclosure thickness is calculated based on all predictable external loads including extreme ones. Since the external enclosure is not designed to be leak-proof, no special seal coating is provided. The shell contains standard reinforcement bars and grids without any pre-stressing. The external enclosure and the plinth rest on the foundation at elevation -4.500.

Internal steel protection enclosure is located above the reactor building plinth, so that it forms, together with the top of the plinth and the reactor concrete cavity, a closed leak-proof space, which constitutes an isolating safety system. Principal safety function of the steel protection enclosure is to prevent further propagation of fission products released as a result of an accident. Its service life is 60 years, including 50 years of reactor operation. Besides, the steel protection enclosure plays an important role in a system of long-term heat removal after accidents involving loss of coolant. The cylindrical part of the enclosure is cooled with a system of closed-loop vertical conduits (on the external surface of the enclosure), filled with water and connected to PHRS tanks, where cooling water in fed from.

The steel protection enclosure is designed to stand internal and external (existing in the annular space) conditions involved in various design basis accidents. The internal enclosure accommodates a number of sealed penetrations for service lines (process and ventilation pipelines) and electric cabling.

5.5.7.4 Turbine building

Turbine building contains power systems and equipment of the secondary circuit. It falls into Safety Category III, and Seismic Resistance Category II. All the premises of the turbine building fall into normally occupied area. All the turbine building buildings are skeleton type, with steel load-bearing elements. The building frame consists of lateral frames including roof stanchions in rows 3 and 4, and roof trusses 36 m in span. The lateral frames are spaced at 12 m.

5.5.7.5 Other buildings

Auxiliary building is one of the most important power unit buildings. It contains systems of primary circuit chemistry control and radwaste cleanup. Auxiliary building is located within the site area exposed to possible contamination. All the auxiliary building premises fall into Safety Category II except a part of basement floor containing boxes with high-active sorbent tanks. Said boxes fall into

Safety Category I. Auxiliary building is adjacent to reactor building. Its footprint is 30×24 m, height of the principal part of the building being 24 m.

Because of functional importance of equipment contained in E&C building, the electric and control (E&C) building falls in into Safety Category I, and Seismic Resistance Category I. Therefore, the internal protective structures are required to stand, apart from conventional loads, also special extreme impacts (aircraft crash, air shock wave, and MDBA-induced seismic load). The E&C building is cast-in-place construction. Local seismic isolation is provided under process and auxiliary equipment important for MCR and SCR operation. It is designed to substantially decrease the impact of aircraft crash seismic load on NPP principal control functions.

Emergency power supply system diesel-generator (EPSS DG) is located in a separate building. Two EPSS DG buildings are provided for each power unit. There is an intermediate diesel fuel depot near each EPSS DG building. The EPSS DG building falls into Safety Category I, and Seismic Resistance Category I. The EPSS DG is installed in a separate building with footprint 9×27 m. Underground tunnel 2.5 m in width is provided adjacent to the short side of a cell along "B" axis. The EPSS DG intermediate fuel depot is located in a separate construction on common basement, immediately adjacent to the communication tunnel. All EPSS DG load-bearing structures and separations are monolithic concrete.

There are two buildings each containing component cooling system and unit standby diesel-generator station provided for every power unit. Each building has 3 cells containing respectively unit standby diesel-generator, auxiliary 0.4 kV switchgear, and component cooling equipment. Component cooling and unit standby diesel-generator building fall into Safety Category I, and Seismic Resistance Category I. All load-bearing structures and separations of the building are monolithic concrete.

The special building is located as close to the reactor buildings as possible, and connected therewith by pedestrian/service line and cabling trestles, as well as special trolley railway for fresh/spent fuel transportation, with a branch to a railway station reloading facility. The building houses four process facilities: fresh fuel storage (FFS), spent fuel storage (SFS), solidified radwaste storage (SRS) with radwaste recycling/compacting unit, and central repair shop (CRS). Fresh fuel storage, spent fuel storage, and highly-active radwaste storage fall into radiation/nuclear Safety Category I. All the rest compartments fall into Safety Category II.

5.5.8 Technical data

General plant data

Power plant output, gross	640	MW(e)
Power plant output, net		MW(e)
Reactor thermal output	1 800	MWt
Power plant efficiency, net	%	
Cooling water temperature	^{o}C	

Nuclear steam supply system

Number of coolant loops	4	
Primary circuit volume, including pressuriser		m^3
Steam flow rate at nominal conditions [4x894 t/h]	993.3	kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	- /7.06	^{o}C/MPa
Feedwater temperature/pressure	223/	^{o}C/MPa

Reactor coolant system

Primary coolant flow rate [14.91m^3/s]	11 000	kg/s
Reactor operating pressure 15.7	MPa	
Coolant inlet temperature, at RPV inlet	293.9	^{o}C
Coolant outlet temperature, at RPV outlet	323.3	^{o}C
Mean temperature rise across core		^{o}C

Reactor core

Active core height	3.53	m
Equivalent core diameter	3.16	m
Heat transfer surface in the core	4 957	m^2
Fuel weight {UO_2]	68.64	t
Average linear heat rate	11.34	kW/m
Average fuel power density	29.75	kW/kg U
Average core power density (volumetric)	65.4	kW/l
Thermal heat flux, F_q	363.1	kW/m^2
Enthalpy rise, F_H		
Fuel material	Sintered UO_2	
Fuel assembly total length	4 670	mm
Rod array	triangle	
Number of fuel assemblies	163	
Number of fuel rods/assembly294		
Number of guide tubes for control rods/instr.	18/1	
Number of spacers		
Enrichment (range) of first core	3.0	Wt%
Enrichment of reload fuel at equilibrium core	3.6	Wt%
Operating cycle length (fuel cycle length)	10	months
Average discharge burnup of fuel	40 400	MWd/t
Cladding tube material	zirconium alloy	
Cladding tube wall thickness	0.61	mm
Outer diameter of fuel rods	9.1	mm
Overall weight of assembly	kg	
Active length of fuel rods	3 530	mm
Burnable absorber, strategy/material		
Number of control rods	121	
Absorber rods per control assembly	18	
Absorber material	B_4C	
Drive mechanism	Magnetic jack	
Positioning rate		mm/min
Soluble neutron absorber	boron acid	

Reactor pressure vessel

Cylindrical shell inner diameter	4 070	mm
Wall thickness of cylindrical shell	190	mm
Total height	19 100	mm
Base material: cylindrical shell	15Kh2NMFA	
RPV head		
Design pressure/temperature	17.65/350	MPa/^{o}C
Transport weight (lower part) [inc. head]	302	t
RPV head		t

Steam generators

Type	horizontal, U-tube heat exchanger	
Number	4	
Heat transfer surface	4 286	m^2
Number of heat exchanger tubes	8442	
Tube dimensions	16×1.5	mm

Maximum outer diameter	4 100	mm
Total height	7 300	mm
Transport weight	300	t
Shell and tube sheet material	10GN2MFA/0Kh18N10T	
Tube material	08Kh18N10T	

Reactor coolant pump

Type	Single-stage, centrifugal pump	
Number	4	
Design pressure/temperature	17.6/350	MPa/°C
Design flow rate (at operating conditions) [3.728m^3/s]	2 750	kg/s
Pump head		
Power demand at coupling, cold/hot	2 600/1 800	kW
Pump casing material	Stainless steel	
Pump speed	1500	rpm

Pressuriser

Total volume	79	m^3
Steam volume: full power/zero power	24	m^3
Design pressure/temperature	17.65/350	MPa/°C
Heating power of the heater rods	2 520	kW
Number of heater rods	28	
Inner diameter	3 000	mm
Total height	13 000	mm
Material	10GN2MFA	
Transport weight	214	t

Pressuriser relief tank

Total volume	m^3	
Design pressure/temperature	MPa/°C	
Inner diameter (vessel)	mm	
Total height	mm	
Material		
Transport weight		t

Primary containment

Type	Dry, double wall	
Overall form (spherical/cyl.)	cylindrical, in steel/concrete	
Dimensions (diameter/height)	41/61.5	m
Free volume, gross	60 000	m^3
Design pressure/temperature (DBEs)	~ 500/	kPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate, inner wall	0.1	vol%/24 h
Is secondary containment provided?	yes, space between the walls	

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage		kV
	rated capacity		MVA
Plant transformers,	rated voltage		kV
	rated capacity		MVA
Start-up transformer	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		MW
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

5.5.9 Measures to enhance economy and maintainability

For expenditures reduction on civil engineering work, simplification (optimization) of the layout and power unit technological approach is envisaged. The results of optimisation of the master plan ensured the area site reducing by 30% due to the following layout solutions:

- Combining of separate buildings of the power unit into common buildings (the fresh fuel unit and laboratory and utility buildings are included to auxiliary building; buildings of emergency diesel-generator stations are built to the building of electrical devices and control etc.). Building of intermediate storage of spent fuel is excluded from the project.
- Laying of cables and pipelines mostly in the earth, laying of communications in tunnels is reduced considerably.

The technological approach applied and the usage of passive safety systems allowed a considerable simplification of the design. In particular:

- Seismic protection requirements to the cooling water system (PE), to the system of intermediate circuit (KAA) and to the system of unit diesel-generators are decreased (from category 1 to category 2). All these systems have safety class 3 and refer to safety important systems. Their operation can be required on any of four levels of protection of physical barriers, at any initial event considered in design, besides the condition with SSE, when safety of power unit is only ensured by complex of passive safety systems;
- Impact on safety requirements to the active system of boron solution injection to the reactor are decreased (from class 2 down to class 3), as far as the system performs its redundant function with respect to passive safety systems at accidents related to primary circuit leakages. The requirements are decreased respectively to backup electrical power supply of equipment of this system, thus allowing the electric power supply from the unit diesel-generators (instead of emergency diesel-generators). Taking into account the fact that the pumps of the active system of boron solution injection made up the main part of load of emergency diesel-generators, the power of the emergency diesel generators is decreased considerably;
- Due to optimization of systems of handling the borated waters, the quantity and capacity of tanks of drain water processing and storage system is reduced.

The construction volume of the I&C building was reduced by 50% due to:

- Application of modern switchgears, enabling to locate up to eighty connections of low voltage in one cabinet of dimensions 2200x1000x1000 mm and to avoid using the current-limiting devices and secondary assemblies of low-powered consumers due to high ability of switching facilities to withstand the short circuit currents. It allowed to cut down the quantity of low voltage cabinets by a factor of three;
- Increasing of single power of auxiliary transformers at a voltage of 10/0.69 and 10/0.4 kV, that allowed to cut down the quantity of transformers by a factor of two. It became possible in connection with application of switching facilities with high ability to withstand the short circuit currents;
- Reducing the I&C cabinets quantity nearly by a factor of two, due to use of the up-to-date digital and optic fiber technologies. The I&C block diagram was also simplified due to this measure;
- Reduction of volume of ventilation systems, due to the above stated reduction of equipment and application of modern equipment with increased reliability, as well as, thereof, reduction of quantity of backup and implification of block diagram;

Application of copper cables with insulation, not containing halogens and not supporting burning, instead of aluminium ones, permits to lower considerably the fire hazard and to cut down expenditures on installation of cable facilities.

Conversion of auxiliary electric power supply to an average voltage of 10 kV instead of 6 kV permits to lower considerably currents of short circuits and, hence, to reduce cross-sections of cables, to lower cost of electrical equipment.
Changes in the technological part allowed to cut down volumes and cost of electrical equipment. They considerably reduced the power of equipment of emergency diesel-electric station. The arrangement of this station is ensured not in separate buildings designed for all external exposure, but in the protected zone of I&C building.

The improvement of VVER-640 NPP design economics is also reached due to the following measures:

- Decrease of cost of design work due to application of standard (base) design and the relevant simplification of the licensing procedure;
- Unification of design approaches with the design of NPP with VVER-1000 and reduction of nomenclature type elements quantity;
- Quantity reduction of equipment, systems and components to be approved by safety authority;
- Reduction of physical volumes due to application of new technologies and modular factory-made components;
- Acceleration of NPP construction;
- Creation of control system for configuration of design during the NPP life cycle;
- Possibility to increase the NPP power (up to 10%) without replacement of main equipment, keeping the incorporated safety criteria.

The estimation of the VVER-640 NPP construction cost was performed on the basis of the approved design for Central Russia climatic area. Calculations of economical indices are performed given the international approach to return of credits within no more than 10 years. The calculations were performed for 7000 hours of effective power operation per year. In fact, this figure can be increased up to 7900 hours and, respectively, the economical NPP indices will increase. Results of calculations of VVER-640 NPP indices are given in Table 5.5-II.

TABLE 5.5-II. RESULTS OF CALCULATIONS OF VVER-640 NPP INDICES

No.	Parameter	NPP with 1 unit	NPP with 2 units	NPP with 3 units
1	Power, MW(e) (gross)	645	1290	1935
2	Electricity generation(million kW*h/year)	4515	9030	13545
3	Electricity supplied to grid (million kW*h/year)	4196	8394	12681
4	NPP construction cost (million US$)	603,3	1117,2	1340,63
5	Specific construction cost (US$/kW)	935,3	866,0	692,8
6	Prime cost of electricity during first 10 years of operation (cent/kW*h), including:	2,43	2,28	1,94
	• capital cost component;	1,44	1,33	1,06
	• fuel component;	0,49	0,49	0,49
	• materials and services;	0,22	0,20	0,16
	• salary;	0,09	0,08	0,07
	• others	0,19	0,18	0,16
7	Prime cost of electricity after 11th year of operation (cent/kW*h)	0,99	0,95	0,88

5.5.10 Project status and planned schedule

For the sites in Sosnovy Bor NPP, Kola NPP-2 and Far East NPP the following was developed: construction feasibility study (FS), preliminary safety assesment report (PSAR), environment impact report (EIR), probabilistic safety analysis (PSA) and technical design (for Kola NPP-2). The design materials have been examined by the state authorities and different expert organisations on regional and federal level, as well as by GRS (Germany) and DOE (USA). The site permissions and license for construction of Kola NPP-2 and Sosnovy Bor NPP and site permission for Far East NPP were received.

Construction of the pilot plant at the Sosnovy Bor site (Leningrad nuclear power station site) is under consideration. This pilot plant will be followed by a number of units at the Kola nuclear power station site and on some other sites in Russia.

References

[1] General safety regulations for nuclear power plants (OPB-88/97).

[2] Nuclear safety rules for reactors of nuclear power plants (PBYa RU AS-89).

5.6 NPP WITH VK-300 BWR (RDIPE, RUSSIAN FEDERATION)

5.6.1 Introduction

The analysis of electric power and heat consumption in the Russian Far East and Siberia shows that these regions (especially those with small power systems) need the medium capacity power sources for heat and electric power generation that are capable of competing economically with organic fuel plants.

Medium power nuclear plant units should meet the specific requirement that their power, governed by the demand, should not exceed the limit ensuring stability of the power system with a sudden shutdown of the unit. Power limitation for autonomous power systems is unlikely to permit the use of a unit with a power of over 250–350 MW(e) in the mix.

A nuclear power plant equipped with a VK-300 boiling water reactor is intended for small- and medium-size power systems as well as for electricity and heat co-generation.

5.6.2 Description of the nuclear systems

5.6.2.1 Primary circuit and its main characteristics

In order to reduce the hydraulic resistance of the coolant circulation circuit, an original coolant circulation and multi-stage separation scheme was selected.

It is generally known that in most of the modern boiling-water reactors with internal steam separation the entire steam-water mixture flow downstream of the core (in VK-300, the average quality at core outlet is 15.5% at feedwater temperature of 190°C) goes to the separators. As a result, the hydraulic resistance of the separators turns out to be high. If moisture is preliminary taken from the flow and returned to the core inlet to reduce the mass flow rate of the steam-water mixture through the separators, it is possible to reduce the hydraulic resistance of the circuit and, as a consequence, increase the natural circulation rate.

The reactor design includes a unit of draft tubes whose functions are:

- Segregation of the coolant flows going up and down;
- Preliminary moisture separation;
- Formation of a water inventory (between the draft tubes) that is immediately returned to the core at reactor shutdown or during accidents;
- To form of a guiding structure for the reactor control rods (which is very important for the upper placement of the control and protection system (CPS) drives).

The possibility of the moisture separation after the steam-water mixture leaves the draft tube unit has been proven experimentally. The cyclone-type steam separators were also experimentally optimized when developing a vertical steam generator concept for WWER-1000.

5.6.2.2 Reactor core and fuel design

The reactor core consists of 313 fuel assemblies (FA), 90 CPS control members, each of which includes three control rods. The control rods are located in central tubes of 270 FAs. The rods of three neighbouring FAs are united as a CPS control member.

The VK-300 reactor uses WWER-1000 fuel elements (fuel element is a rod with a cladding of E110 zirconium-based alloy). The uranium enrichment is up to 4 %. The core height is 2420 mm.

5.6.2.3 Fuel handling and transfer systems

A special refuelling machine (its telescopic rod) is used to withdraw the FAs from the reactor and reshuffle them beneath the water level in the core to provide for the radial neutron flux flattening. FAs are withdrawn and reshuffled together with the control rod. The fuel cladding is checked for leaktightness immediately in the refuelling machine rod during this operation. Following the withdrawal, the FAs are placed in the spent fuel pool where they are stored for 6 years and are subsequently delivered for reprocessing in shipping casks.

The reactor internals are withdrawn for periodic (once in 4 years) inspection and visual examination by a gantry also beneath the water level. The state of this equipment is checked remotely in special vaults.

5.6.2.4 Primary components

The reactor vessel is made of 12Kh2NMFA pearlite heat-resistant steel and its internal surface is lined with austenite corrosion-resistant steel.

The key internals are installed in a cylindrical vault and are fastened to the vessel near its flange. The vault bottom accommodates fuel assemblies containing movable control members. The draft tubes are arranged as a removable unit above the core. Above the draft tubes there is a separation chamber, in which steam is largely separated from the water that is reversed and flows down to the core inlet over the annular downcomer channel between the core and the vessel.

The steam-water mixture from the separation chamber goes to cyclone separators (also arranged as a removable unit). After it leaves the separators, the steam with the humidity of 0.1% goes to the turbine through a pipeline.

The reactor vessel has two headers – a water header and a steam header. When in the water annular header, the feedwater is distributed uniformly throughout the peripheral reactor section and goes to the core inlet as it mixes with the water flowing down from the separation chamber.

The steam header is an annular enclosure connecting the steam space of the reactor vessel to the steam nozzles.

The drives of the CPS control members are installed on the reactor vessel cover. Prior to the cover removal, tie-rods of control members are separated from electric motors.

The reactor facility design uses innovative approaches based on the previous experience of designing and operating similar nuclear reactors. Thus, a VK-50 reactor operated in Russia is used as a prototype for the VK-300 reactor. The experience of the design and long-term operation of the boiling-water small-size VK-50 reactor in Dimitrovgrad turned out to be extremely useful in the development of the VK-300 reactor. International achievements in the area of designing and operating boiling-water reactors, especially with respect to the design of passive safety systems, have also been taken into account in the design of VK-300 reactor.

The VK-300 design uses the basic equipment developed and manufactured for reactors of other types. Thus, the VK-300 design uses the WWER-1000 reactor vessel. It is evident that it is difficult, time-consuming and expensive to design and launch into production a new pressure vessel for a power reactor. So using in the design an already completed development of a nuclear reactor vessel considerably facilitates and simplifies the task of designing the VK-300 reactor. Production facilities for manufacturing such pressure vessels have been built in Russia. Besides, there are already over 10 manufactured vessels and it is an economically expedient task to recover them in national economy.

Therefore, the basic equipment for the innovative boiling-water VK-300 reactor has been well developed and has an extensive experience of operation.

5.6.2.5 Reactor auxiliary systems

No information provided

5.6 2.6 Operating characteristics

Reactivity effects and coefficients that set up the basis for the reliable controllability of the reactor and its stable operation are very significant for the successful performance of the control function of the fission chain reaction. The VK-300 reactor has just a small reactivity margin for burnup thanks to partial refuelling and the use of the burnable poison. The minimization of the reactivity margin ensures prerequisites for designing a simplified CPS system with "light" rods that mitigates the consequences of an accident with self-withdrawal of the CPS rods.

The reactor has two reactivity control systems using different principles of operation. The first of the systems is a traditional rod system including 90 drives of the CPS actuators. Each of the drives simultaneously moves the control rods in the three adjoining core fuel assemblies.

The second reactivity control system is a liquid system intended for introducing the boric acid solution to the reactor coolant during accidents or at failures of the reactivity control rod system. The system consists of pressurized hydraulic accumulators with the boric acid solution.

5.6.3 Description of the turbine generator plant systems

No information provided

5.6.4 Instrumentation and control systems

5.6.4.1 Design concept, including control room

The computer-aided process control systems of a power unit with a VK-300 reactor facility (RF) is intended for:

- Cost effective and reliable electricity and heat generation;
- Observance of the operational limits established in the design for normal operation;
- Observance of the safe operation limits and conditions to prevent the progression of anticipated operational events to emergency situations and accidents;
- Limitation of the consequences of design-basis and beyond-design-basis accidents.

The system includes subsystems performing the monitoring and diagnosis functions with respect to all equipment of the power unit:

- A metal diagnosis system;
- A radiation safety monitoring system.

The following systems that integrate the above subsystems as a complex system are considered as the upper level of the system's hierarchic structure:

- An operating personnel information support system;
- The main control panel;
- The backup control panel.

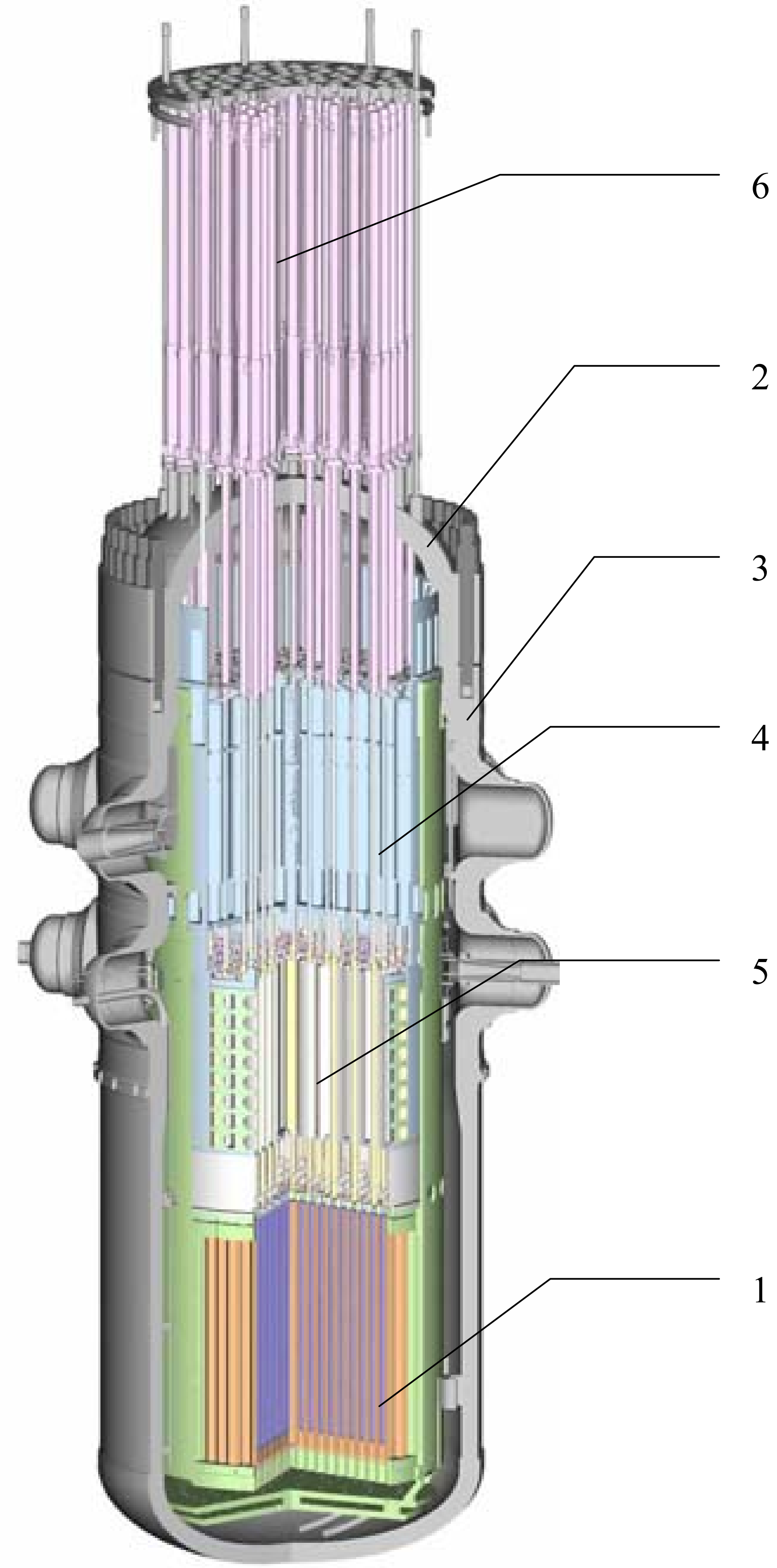

FIG. 5.6-1. VK-300 reactor.
1 - core; 2 – reactor cover; 3 – reactor vessel; 4 – separators;5 – draft tubes; 6 – CPS drives

5.6.4.2 Reactor protection and other safety systems

The control and protection system is intended for monitoring the relative power level, power and period rate change, reactivity as well as for controlling reactivity and reactor power in different modes of the reactor operation.

The CPS provides for:

- Manual remote control of working members;
- Power monitoring in the range of (10^{-7} –120) % N_{nom};
- Monitoring of the steady-state power change period in the limits from 120 to 10 s in the power change rage of (10^{-7} – 120) % N_{nom};
- Monitoring of the power change rate in the limits of (0-10) % N_{nom}/s in the power range of (10^{-7} –120) % N_{nom};
- Reactivity monitoring in the limits of (0-0.7) β_{eff} in the power range of (10^{-7}–120) % N_{nom} and monitoring of the reactivity change rate in the limits of (0-0.3) β_{eff}/s in the power range of (10^{-2} – 120) % N_{nom};
- Manually remotely bringing the reactor from a subcritical state to the specified power level in the range of (1-100) % N_{nom} and keeping the specified power level using manual control means;
- Automatic startup of the reactor from a subcritical state with control for reactivity and bringing to the power level of 1 % N_{nom};
- Automatically changing and automatically keeping the specified power in the range of (1-100) % N_{nom}.

The following safety assurance principles were met in the CPS design:

1. Availability of two reactor shutdown systems, each of which is capable of ensuring the reactor core transfer to a subcritical state independently of the other.

2. Defence-in-depth consisting of five barriers:
   - Warning alarm – providing the operator with information on the beginning of a deviation from the normal operating conditions;
   - Interlocks that limit the development of situations with potential unauthorized reactor power increase;
   - Fast controlled power reduction using an automatic controller;
   - Automatic power reduction to 10% N_{nom} by the automatic controller;
   - Reactor scram.

3. Use of two spatially and functionally independent hardware sets for formation of emergency signals.

The reactor power is controlled using 90 working members (WM), six of which are used for automatic control. The remaining WM are divided into scram working members (SWM) and manual control working members (MCWM). Structurally, each working member is a bundle of three rods moving in three neighbouring and equally distant FAs inside the central tubes of 61×1 mm. The control rod position is central. The working members are withdrawn from the reactor upwards. The maximum WM stroke is 2600 mm.

The maximum efficiency of one working member is 2.2% for the minimum-controlled power level and 0.4% at the working parameters.

The effective delayed neutron fraction β_{eff} for the initial load is 0.75 %, and 0.56-0.61% in the equilibrium refuelling mode.

The movement rate of each WM in the manual mode is (40±5) mm/s or not more than 0.01 β_{eff}/s.

An electric drive with a linear stepped motor is used as the actuator drive. The WM is inserted into the core by gravity during a loss of power at the stepped motor windings.

The position of each CPS WM is monitored using an individual induction position sensor being part of the actuator.

5.6.5 Electrical systems

No information provided.

5.6.6 Safety concept

5.6.6.1 Safety requirements and design philosophy

The validated NPP personnel exposure is established by regulatory documents effective in Russia.

The observance of the radiation safety criteria is ensured if no design limits (operational limits and safe operation limits) are exceeded in terms of:

- the primary water activity level;
- the grid water activity level;
- the activity level of the water that cools the system's equipment adjoining the primary circuit as well as the allowable liquid waste;
- the allowable release of radioactive substances to the atmosphere via the stack;
- the allowable radiation level in the NPP rooms.

The reactor facility design has established the design limits for the RF systems (components), the observance of which will ensure the observance of the radiation safety criteria. The main design limits are those for fuel elements and associated reactor coolant radioactivity levels in terms of reference isotopes.

Non-excess of the standards set by PBYa RU AS-89 for WWER and nuclear district heating plant (NDHP) fuel elements has been assumed as the maximum design limits for the fuel element damage, namely: the fuel cladding temperature is not more than 1200 °C, the local cladding oxidation depth is not more than 18 % of the initial wall thickness and the reacted zirconium fraction is not more than 1 % of its mass in the fuel claddings.

The maximum design level of the fuel element damage is set for the most severe accidents. In terms of the cladding temperature and the local cladding oxidation depth, the admissibility of using this limit for the VK-300 reactor fuel elements is shown in the design. As to the reacted zirconium fraction, estimates show that the burning of 1% of the fuel cladding zirconium mass can generate 290 m^3 of hydrogen. According to the very conservative estimate with no regard for the operation of the system for hydrogen removal from the primary containment (PC) and removal of non-condensed gases from the PC and the emergency cooldown tank to the activity suppression facility at such hydrogen release and the PC volume of 1725 m^3, its concentration in the steam-gas mixture (~9%) is not enough for detonation.

The allowable level of the reactor coolant activity is determined by the safe operation limit in terms of the quantity and magnitude of fuel element defects. The limit set by PBYa RU AS-89 for the NDHP (0.1% of fuel elements with gas leakage defects and 0.01% of fuel elements with a direct coolant and nuclear fuel contact) has been assumed as the safe operation limit in terms of the quantity and magnitude of fuel element defects. The coolant activity in the course of operation is monitored by the fuel cladding failure detection system (CFDS) and, if the safe operation limit for the activity of the reference nuclides is exceeded, the reactor operation should be suspended for search of defective fuel elements and restoring normal operation conditions.

The operational limit for the quantity and magnitude of defects and the respective activity values representative of the normal operating conditions are 1/5 of the safe operation limit.

The operational limit for the fuel cladding external surface temperature is 400 °C and 500 °C for the fuel cladding internal surface temperature.

In carrying out a computational analysis of accidents, the assessment of their consequences also uses the criteria allowing the evaluation of the fuel cladding state, i.e. fuel cladding integrity and respectively the maintenance of the safety barrier functionality. Non-excess of these criteria guarantees the integrity of the safety barrier.

Considered as such criteria were:

- the fuel cladding temperature limit,
- the peak radial-average fuel enthalpy,
- the maximum fuel temperature.

The fuel cladding temperature limit that ensures the cladding integrity at a thermal-mechanical cladding – fuel interaction has been assumed as equal to 700°C. The energy released during a short period of time in a nuclear fuel mass unit at a fast reactivity introduction is considered as the specific threshold energy of the fuel element damage. The peak radial-average fuel enthalpy at any core point should not exceed 230 cal/g.

The maximum fuel temperature is limited by the melting temperature, which is assumed to be equal to 2800 °C with regard for burnup.

The designs of normal operation systems and safety systems were developed such that the estimated values of the probability of a ultimate source term and the total probability of severe beyond design-basis accidents (with the fuel element damage in excess of the maximum design level) were at as low as reasonably achievable (technically expedient level) and did not exceed the values set by OPB-88/97 (10^{-7} and 10^{-5} per reactor-year respectively). There are technical means for control of beyond design-basis accidents for mitigating their consequences.

The RF safety is ensured by a system of technical and organizational measures including:

- The consistent implementation of the defence-in-depth concept;
- The use and development of the reactor inherent safety features;
- Organization of the safety system operation based on the principles of:
  - redundancy,
  - spatial and functional independence,
  - single failure,
  - diversity.

The basic VK-300 RF safety assurance principle is the consistent implementation of the defence-in-depth concept based on using the system of physical barriers against the spreading of ionizing radiation and radioactive substances to the environment, and a multi-level system of technical and organizational measures for protection of barriers and preservation of their efficiency. The system of physical barriers includes:

- Fuel matrix;
- Fuel cladding;

- Reactor vessel, primary circuit pipelines, residual heat removal system pipelines and components;
- Primary containment, emergency cooldown tanks;
- Leaktight rooms of the reactor building.

The analysis has shown high safety of the reactor facility. The probability of a severe core damage will not exceed $1.4 \cdot 10^{-7}$ per reactor-year.

5.6.6.2 Safety systems and features (active, passive and inherent)

The use of the Primary Containment (PC) is an economic and reliable approach to the safety assurance problem thanks to the use of structurally simple passive safety systems. The PC has a small volume (about 1700 m^3). It performs the functions:

- As a safeguard (additional) reactor vessel;
- As a protective safety barrier that confines radioactive materials within its boundaries during accidents with ruptures of steam, feed water and other pipelines immediately near the reactor;
- Of enabling emergency cooling of the reactor with the reactor coolant without the need of an additional coolant inventory.

Located outside the PC are Emergency Cooldown Tanks (ECTs) intended for:

- Accumulation of the reactor energy (ensuring the possibility of its transfer to the ultimate heat sink for an unlimited period of time);
- Replenishment of the cooling water inventory in the reactor during accidents through the return of the condensed coolant to the reactor.

A simplified hydraulic scheme of the PC as a set with the ECTs is shown in Fig. 5.6-2.

The pressure in the reactor and in the PC is levelled rapidly during ruptures of steam or feed water pipelines inside the PC. It creates conditions for the water inflow from the ECTs to the core via a special pipeline. The initial accident stage proceeds safely without reactor makeup as the water inventory in the reactor is enough to ensure normal heat removal from the core. Later, as the pressure decreases in the reactor and the pressures in the reactor and in the PC are levelled, the water goes to the reactor from the ECTs by gravity. An external circulation circuit is formed - ECT, reactor, PC, ECT. The water from the ECTs is accumulated in the PC with time but it does not affect the serviceability of the tanks as the water inventory therein is enough for filling up the PC volume and successful operation of the "external" natural circulation circuit. It should be noted that the PC is automatically pressurized during accidents with ruptures (using special passive action valves) to exclude releases of the radioactive coolant beyond the PC.

Another class of accidents includes accidents with the loss of heat removal from the reactor due to a turbine failure or accidents in the reactor's external feed water line. The major task here is to receive heat from the reactor and ensure its normal cooldown. This is ensured by a special system for passive heat removal from the reactor based on the use of steam condensers located in the PC around the reactor. These condensers are connected with the reactor through pipelines that are flooded with the primary circuit water during normal operation of the reactor. When the water level decreases in the reactor, the upper pipeline is opened to let the steam from the reactor to the condensers and the condensate flows back to the reactor. The condensers as such are cooled with water from the emergency cooldown tanks. The system is based on a fully passive principle and intended for natural heat transport from the reactor to the emergency cooldown tanks.

It should be noted that the emergency cooldown tanks are also intended for receiving the discharge from the reactor safety valves located inside the PC.

The above examples show that the heat from the reactor is accumulated in the emergency cooldown tanks. The heat capacity of the tanks as such is enough for independent operation throughout the day (i.e. without personnel interference). This interval may be prolonged for an infinite period of time thanks to the operation of the heat removal system from the tanks to the ultimate heat sink. This is a simple and reliable system consisting of two heat exchangers connected with pipelines. One of the heat exchangers is plunged into the emergency cooldown tank water and the other is installed in the atmospheric airflow outside the reactor hall. The coolant in the system is water circulating in the circuit naturally without circulation boosters.

5.6.6.3 Severe accidents (beyond design basis accidents)

Severe accidents are unlikely for this reactor type due to the low energy intensity of the core and the availability of passive cooldown systems. However, based on a deterministic approach to the potentiality of a severe accident, under consideration are such means of reducing its consequences by confinement of the melt within the reactor vessel and by localization of radioactive products in the primary containment and the containment.

5.6.7 Plant layout

5.6.7.1 Buildings and structures, including plot plan

No information provided

5.6.7.2 Reactor building

No information provided

5.6.7.3 Containment

Despite a high safety level of the plant shown by the computational analysis, a decision was made (with regard to the assumed plant deployment near cities for their heating) to install under the containment not only the nuclear island but also the steam-turbine unit. This enables protection of the nuclear island from aircraft crash and prevention of the radioactive coolant release to the atmosphere during depressurization of the primary circuit pipelines beyond the PC with failure of isolation gate valves on the PC boundary to close (the probability of 10^{-9} per reactor - year).

5.6.7.4 Turbine building

No information provided.

5.6.7.5 Other buildings

No information provided.

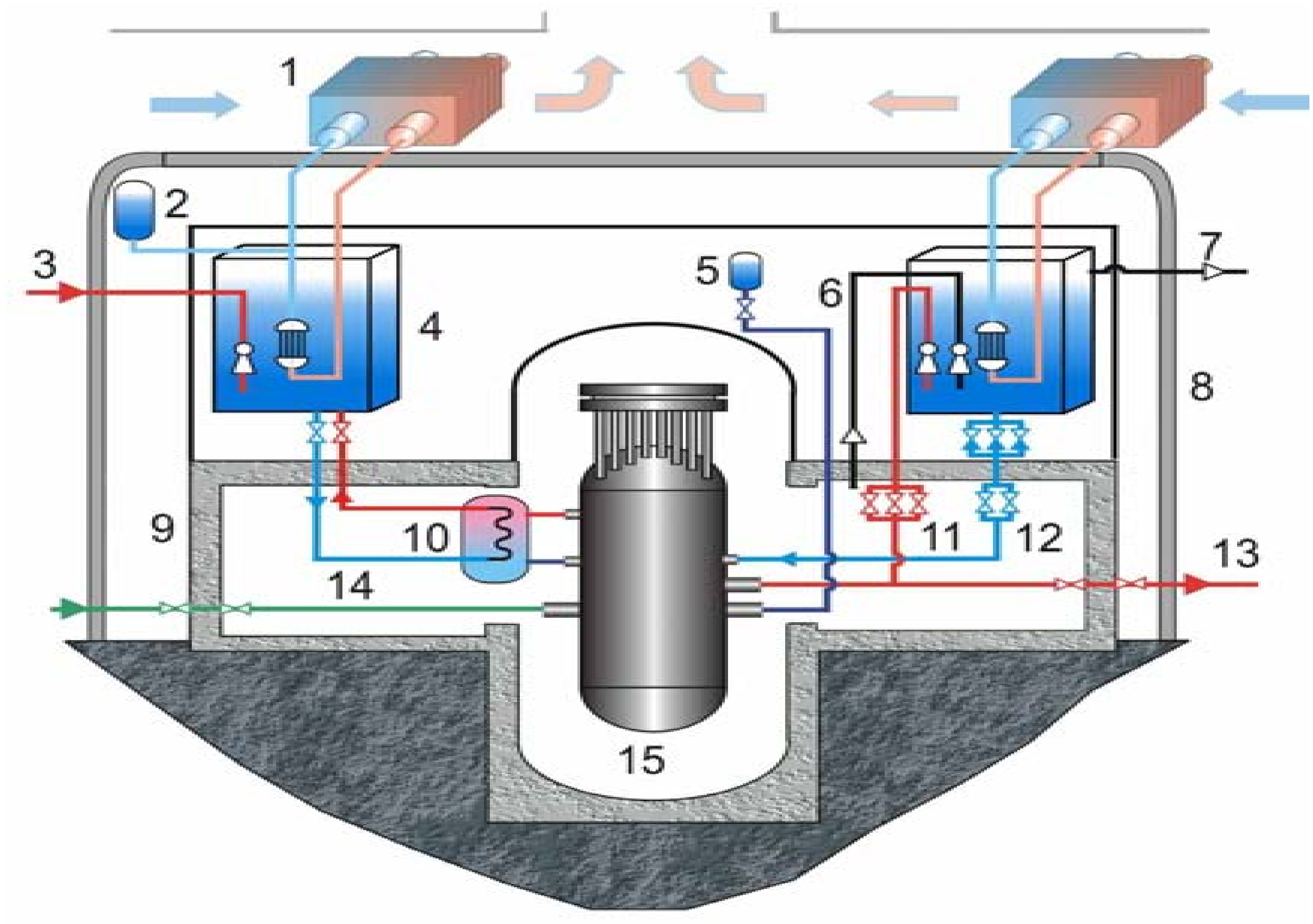

FIG. 5.6-2. Simplified hydraulic diagram of the VK-300 reactor facility.

1- air heat exchanger of RHRS to ultimate heat sink; 2 – pressurizer; 3 - emergency turbine hall steam discharge; 4 - emergency cooldown tank; 5 - liquid shutdown system vessel; 6 - emergency PC steam discharge; 7- to the activity suppression facility; 8 – containment; 9 – PC; 10 - RHRS condenser; 11 - main safety valve; 12 - emergency core cooling system; 13 - steam outlet; 14 - feedwater inlet; 15 – reactor

5.6.8 Technical data

The main characteristics of a two-unit NPP design with VK-300 reactors are as follows:

Number of reactor units	2
Thermal power, MW	750×2
Heat generation capacity, Gcal/h	400×2
Electric power, MW	
under district heating mode	150×2
under condensation mode	250×2
Steam parameters at the reactor outlet	
– pressure, MPa	6.86
– temperature °C	285
– moisture content %	0.1
Reactor steam output, t/h	1360
Number of hours of using installed power per year	7000
Number of hours of using nominal power for heat generation per year	5600
Uranium load, t	32.2×2
Fuel enrichment, %	4
Fuel burnup, MWd/kgU	42.4
Gross efficiency	
– in district heating mode	0.706
– in condensation mode	0.333

5.6.9 Measures to enhance economy and maintainability

The concept of maximum simplifications of the design of main equipment and systems is laid as the basis of the design for the power unit with the VK-300 reactor facility. A one-circuit diagram with natural circulation of coolant and integrated layout is realized. The effective flow chart of circulation and multistage separation of the two-phase coolant in reactor is applied. Passive safety systems have been developed for the VK-300 reactor facility placed inside the primary containment.

Significant margins for specific core power and separation loads of in-core separation devices were introduced for physical and thermal-hydraulic calculations of VK-300. After start-up of the first unit of nuclear district heating and electric plant (NDHEP) with reactor VK-300 and trial tests these margins can be reduced. This will allow increasing the reactor thermal power and, thus, the electrical power of the power unit. The margins for capacities of valves and turbine high-pressure cylinder have been envisaged for this purpose.

The elaboration of core, which will be refuelled once every two years, is envisaged to reduce the scope of work with spent fuel. Transition to FA without zirconium will allow to prolong the fuel cycle and to increase the time between refuelling. These measures will provide availability factor increase and reduction in the scope of maintenance.

Further, it is possible to convert to "one time per cycle" refuelling, when all the core is taken out completely and is loaded into a special container, which is loaded in a special building for spent fuel storage. This solution allows to place the spent fuel pool outside of containment and to reduce its size.

To rule out a release of radioactive products existing in the steam of single-circuit power unit, at normal and accident conditions, the reactor and turbine installation are placed on a shared bed plate inside common containment made of reinforced-concrete, not pre-stressed, which is also intended for protection against external impacts.

Low-pressure heaters 3 and 4, as well as low and high pressure boilers of the intermediate circuit (for the district heating system) were developed and combined in one vessel to reduce the number of units of equipment, make lower its cost and reduce the space occupied by the turbine unit.

The turbine is developed with a high pressure separator located downstream the groups of the turbine high-pressure stages. The steam from this separator comes into the group of medium-pressure stages. Due to this solution, the average humidity in the flow path of the high-pressure cylinder is reduced not less than by 3%, increasing the internal efficiency of the turbine and reducing the erosion of the steam flow path.

To simplify the removal of spent fuel and taking away large-size components of equipment for repair, a railway entrance is made in the containment, which can be served by a polar crane.

System of power distribution is made from sulphur hexafluoride current distributors and package switchgears located in closed channels and rooms.

To reduce the repair work, the surface of condensers is made from titanium tubes. Their service life in practice is equal to the service life of power unit.

All the industrial and household effluents, and the de-contaminated waters from the site are used in the work cycle of NDHEP, i.e. no discharge into natural water basins is foreseen.

The heat circuit and plant control system are executed in such a manner that at turbine trip or switching-off the generator from the grid no heat supply termination occurs due to steam supply into the boiler of high pressure intermediate circuit directly from the reactor through pressure-reducing device. This solution increases the availability factor of the heat-supplying system.

To increase the plant site vitality, a backup electric power station is envisaged that has diesel-generators and steam boilers that provide electric and thermal power supply to station auxiliaries at loss of off-site and on-site power.

An important measure reducing the capital costs and construction time will be the usage of the floating method of building the main structures of NDHEP. With this, the main building of the power unit including the containment, reactor and turbine is erected on a barge (pontoon) in the shops of a shipbuilding plant. This complex is to be delivered via waterways and specially dug channels to the site and installed on a specially prepared basement. Then the channels are filled up, and water is pumped out of the basement pool. This method provides a high quality of construction and mounting activities in the main building and the reduction of NDHEP construction time.

To rule out the operation of NDHEP at partial loads and to increase the availability factor, grid water accumulator (GWA) is designed to be built near the NDHEP. This will allow to load to the maximum the heat-supplying equipment in winter at night (when electric power consumption goes down) by pumping the heated grid water into the GWA. At the daytime, the grid water from the GWA will be used to load the turbine to the maximum. In summer, when heat is generated only for hot water supply, it is possible at night to fill up the GWA with grid water, and at daytime to switch over the turbines to condensation mode, which will provide generation of the maximum quantity of electric power.

5.6.10 Project status and planned schedule

The VK-300 RF detailed design was completed in 2000. Works have been planned and are underway for the experimental substantiation of the reactor circulation circuit as required to reduce excessive conservatism that was laid in the reactor calculations. Besides, experimental and design work is required for the CPS rod drives for which purpose full-scale drives, bars and CPS rod coupling devices are manufactured to be used in bench tests. The entire R&D complex is proposed to be completed during 2-3 years after which the reactor contractor design with the R&D results taken into account will be issued.

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5.7 IRIS (INTERNATIONAL CONSORTIUM LED BY WESTINGHOUSE, USA)

5.7.1 Introduction

IRIS (International Reactor Innovative and Secure) is a modular, integral-type pressurized light water cooled, medium power (1000 MW(t)) reactor. The IRIS development program was originally sponsored by the U.S. Department of Energy (DOE) as part of the NERI (Nuclear Energy Research Initiative) program. IRIS was selected as an International Near Term Deployment (INTD) reactor, within the Generation IV International Forum activities. The IRIS concept addresses the top-requirements defined by the U.S. DOE for next generation reactors, i.e. enhanced reliability and safety, proliferation resistance, and improved economics. IRIS is an advanced design that does not require new technology development, since it relies on proven light water reactor technology. IRIS is being developed by an international consortium led by Westinghouse Electric Company; the consortium includes a number of U.S. and international companies, universities, national laboratories and organizations, listed in Table 5.7-I.

TABLE 5.7-I. IRIS CONSORTIUM

Industry		
Westinghouse	USA	Overall coordination, core design, licensing
BNFL	UK	Fuel and fuel cycle
Ansaldo Energia	Italy	Steam generators design
Ansaldo Camozzi	Italy	Steam generators, CRDMs fabrication
ENSA	Spain	Pressure vessel and internals
Curtiss-Wright	USA	Pumps
NUCLEP	Brazil	Containment, pressurizer
Bechtel	USA	BOP, AE
OKBM	Russia	Testing
INB (pending)	Brazil	Fuel assembly design
Laboratories		
ORNL	USA	I&C, PRA, core analyses, shielding, pressurizer
CNEN	Brazil	Pressurizer design, transient and safety analyses
ININ	Mexico	PRA support
Universities		
Polytechnic of Milan	Italy	Safety analyses, shielding, thermal hydraulics, steam generators design, advanced control system
MIT	USA	Advanced cores, maintenance
Tokyo Inst. of Technology	Japan	Advanced cores, PRA
University of Zagreb	Croatia	Neutronics, safety analyses
University of Pisa	Italy	Containment analyses
Polytechnic of Turin	Italy	Human factors, reliability availability maintainability
University of Rome	Italy	Radwaste system, occupational doses
Power Producers		
TVA	USA	Maintenance, utility prospective
Eletronuclear	Brazil	Developing country utility prospective
Associated US Universities (NERI programs)		
U. California Berkeley	USA	Neutronics, advanced cores
U. of Tennessee	USA	Modularization, I&C
Ohio State	USA	In-core power monitor, advanced diagnostics
Iowa State (Ames Lab)	USA	On-line monitoring
U. of Michigan (& Sandia Lab)	USA	Monitoring and control

The IRIS design features an integral reactor vessel that contains all the reactor coolant system components, including the pressurizer, steam generators, and reactor coolant pumps. This integral reactor vessel configuration allows the use of a small, high design pressure, spherical steel containment resulting in a high level of safety and economic attractiveness. The IRIS reactor development has employed a “safety by design” approach that has eliminated or reduced the consequences of most accident sequences. IRIS fuel assemblies have standard Westinghouse features but can operate over a three- to four-year long fuel cycle. In order to take advantage of the extended IRIS fuel cycle and to improve the overall plant availability, an optimized maintenance approach for all major components is being developed, which will also extend the interval between maintenance shutdowns to as long as 48 months.

The IRIS design builds on the proven technology provided by 40 years of operating PWR experience, and on the established use of passive safety systems pioneered by Westinghouse in the NRC certified AP600 plant design. The IRIS safety by design (by eliminating or reducing the consequence of accident initiators) provides a very effective first level of defense in depth. In addition, the safety by design is complemented by the adoption of passive safety systems, which, like in the AP600 and the AP1000 designs, once actuated, rely only on natural driving forces such as gravity and natural circulation flow for their continued function. Because of the safety by design approach, the number and complexity of the safety systems and required operator actions are minimized in IRIS versus the passive loop PWRs. The net result is a design with significantly reduced complexity and improved operability, and extensive plant simplifications to enhance construction.

IRIS is being designed to comply with all applicable U.S. NRC criteria. Safety analyses and a Probabilistic Risk Analysis (PRA) are in progress and a testing program is being developed. The preliminary PRA results show a very low core damage frequency that exceeds the goals established for advanced reactor designs, as well as a very low frequency of release due to an innovative reactor and containment cooling strategy. This will allow IRIS to eliminate or at least reduce the need for off-site emergency response. The simplified safety systems reduce surveillance requirements by enabling significantly simplified technical specifications. Built-in testing capability is provided for critical components.

The IRIS plant layout will ensure adequate access for inspection and maintenance. Laydown space for staging of equipment and personnel, equipment removal paths, and space to accommodate remotely operated service equipment and mobile units are part of the plant design. Access platforms and lifting devices are provided at key locations, as are service provisions such as electrical power, demineralized water, breathing and service air, ventilation and lighting. The integral configuration, extended fuel cycle and extended maintenance intervals allow IRIS to significantly reduce the workers exposure and therefore readily adhere to the ALARA requirement.

IRIS is designed with environmental consideration as a priority. The safety of the public and the power plant workers, and the impact on the environment have all been addressed as specific design goals:

- Operational releases have been minimized by design features;
- Aggressive goals for worker radiation exposure have been set;
- Total radwaste volumes have been minimized;
- Other hazardous waste (non-radioactive) has been minimized.

The IRIS design is taking advantage of previous design efforts made in the ALWR program that developed the AP600 plant design and others. This includes use of the Utility Requirements Document (URD) for ALWRs developed by Electric Power Research Institute (EPRI) with a broad participation of numerous countries. The URD takes into account the wealth of information related to nuclear power plant safety and operations that has been generated worldwide with commercial nuclear power, delineates utility desires for the next generation of nuclear plants, and consists of a comprehensive set of design requirements.

Some of the high-level design characteristics and goals of the IRIS plant are:

- Net electrical power of approximately 335 MW(e); and a thermal power of 1000 MW(t);
- Robust core design with an extended fuel cycle length and at least a 15% operating margin on

core power parameters (i.e., to DNBR);

- Short lead time (four years from owner's commitment to commercial operation) and construction schedule (two years);
- No plant prototype is needed since the power generating system components are well based in current technology and will be extensively tested before NRC design certification;
- A significant number of accident initiators is either eliminated outright or their consequences/ probabilities are reduced by design, i.e., without any need of active or passive systems (safety-by-design);
- Major safety systems are passive; they require no operator action or off-site assistance for 1 week after an accident, and additional core and containment cooling is provided for a protracted time without AC power;
- Predicted core damage and release frequency are less than $1x10^{-07}$/yr and $1x10^{-08}$/yr, respectively, and are significantly less than the NRC $1x10^{-05}$/yr and $1x10^{-06}$/yr requirements;
- Reduced, and as a goal eliminated, need for off-site emergency response;
- The core design is capable of operating on a 4-year fuel cycle;
- Refueling and maintenance outages will be significantly less frequent than the current outages;
- Occupational radiation exposure expected to be well below 0.5 man-Sv/yr (50 man-rem/yr), due to the integral configuration and infrequent reloading/outages;
- Plant design life of at least 60 years without replacement of the reactor vessel;

5.7.2 Description of the nuclear systems

5.7.2.1 Primary circuit and its main characteristics

IRIS is a pressurized water reactor featuring an integral primary circuit layout instead of the typical PWR loop layout. All the main primary system components (core, pressurizer, reactor coolant pumps and steam generators) are located inside the reactor pressure vessel, as shown in Figure 5.7-1(a). The primary coolant flow path is illustrated in Fig. 5.7-1(b). Water flows upwards through the core and upward through the riser region (defined by the extended core barrel). At the top of the riser, the coolant is directed into the upper annular plenum where the suction of the reactor coolant pumps is located.

Eight pumps are employed, and the flow of each pump is directed downward through its associated helical coil steam generator module. The flow path continues down through the annular downcomer region outside the core to the lower plenum and then back to the core completing the primary coolant flow path.

The integral primary circuit layout eliminates primary piping outside the pressure vessel and large primary vessel penetrations, thus eliminating the possibility of large break loss of coolant events. It is amenable to an overall reduction of other piping, thus reducing the probability of occurrence for small break loss of coolant events. The integral reactor coolant system pressure boundary provides a barrier against the release of radioactivity generated within the reactor and is designed to provide a high degree of integrity throughout operation of the plant.

5.7.2.2 Reactor core and fuel design

The IRIS core and fuel characteristics are similar to those of a conventional Westinghouse PWR design. However, several features have been modified to enhance performance as compared to conventional plants, while retaining existing technology. An IRIS fuel assembly consists of 264 fuel rods in a 17x17 square array. The central position is reserved for in-core instrumentation, while the remaining 24 positions have guide thimbles. The IRIS fuel assembly design is similar to the Westinghouse 17x17 XL Robust Fuel Assembly design and AP1000 fuel assembly design. Low-power density is achieved by employing a core configuration consisting of 89 fuel assemblies (shown in Fig. 5.7-2) with a 14-foot (4,267 mm) active fuel height, and a nominal thermal power of 1,000 MW(t).

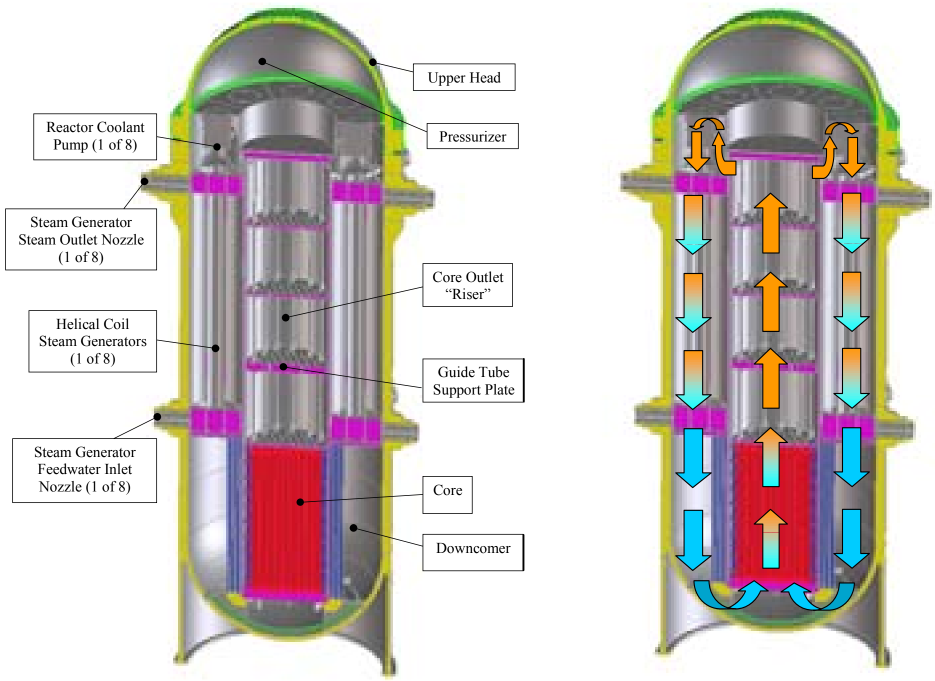


FIG. 5.7-1. IRIS integral layout: (a) main components; (b) main flow path

This results in reduction of the average linear power density by about 25 percent as compared to AP600. The improved thermal margin provides increased operational flexibility, while enabling longer fuel cycles and increased overall plant capacity factors.

The IRIS fuel employs a lattice with enhanced moderation that results in increased discharge burnup, by up to 9%. Another feature contributing to lowering fuel cycle cost and extending reactor life is the use of a stainless steel radial neutron reflector (Fig. 5.7-3). This reflector reduces neutron leakage thereby improving core neutron utilization, extending fuel cycle and further increasing discharge burnup, by ~3%. The radial reflector has the added benefits of reducing the fast neutron fluence on the core barrel and reactor vessel, resulting in a cold vessel (potentially disposable of as non-radioactive material), and reducing the activation of materials and workers exposure outside the vessel.

Reactivity control is achieved in a traditional manner by a combined use of soluble boron, integral absorbers, and control rods. However, soluble boron concentration is reduced as compared to conventional PWR cycles, to improve core response in transients (more negative reactivity coefficients) and reduce the amount of waste to be processed. Another core design feature (common with the AP600 and AP1000 design) is the use of reduced-worth control rods ("gray" rods) to achieve daily load follow while minimizing the required change in the soluble boron concentration. With the exception of the neutron absorber materials used, the design of the gray rod assembly is identical to that of a normal control rod assembly.

Several reloading strategies are available depending on the utility requirements and priorities. When the cycle length is the primary objective, straight-burn core design utilizing enrichment close to 5% can provide a four-year cycle lifetime with a burnup of ~40,000 MWd/tU. The use of erbium integral burnable absorber ensures adequate reactivity control while maintaining a negative temperature coefficient of reactivity. A more traditional multi-batch reloading enables achieving average batch discharge burnup of ~50,000 MWd/tU (for a two-batch reload scheme), or up to ~60,000 MWd/tU (for three-batch reload scheme).

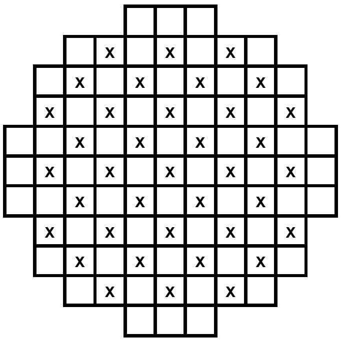


FIG. 5.7-2. IRIS core configuration and a typical RCCA pattern

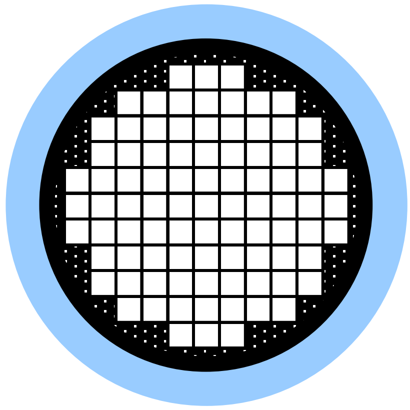


FIG. 5.7-3. IRIS radial neutron reflector assembly

The two-batch reload is compatible with the currently U.S. NRC licensed maximum allowed burnup (62,000 MWd/tU lead rod average) and is therefore the current reference core design. The three-batch reloading scheme may be implemented in the future to improve fuel economy, when fuel with higher allowed discharge burnup (e.g., 62/75,000 MWd/tU batch and lead rod average) becomes licensed. Moreover, the IRIS core was designed to facilitate future upgrade and transition to an eight-year cycle (possibly with a short maintenance shut-down at mid-cycle) by using 8-10% enriched UO_2 fuel or MOX fuel with 10-12% Pu.

5.7.2.3 Fuel handling and transfer systems

The IRIS reactor vessel (RV) is contained in a spherical, steel containment vessel (CV) that is 25 meters (82') in diameter (Figure 5.7-4). The containment vessel has a bolted and flanged closure head at the top that provides access to the RV upper head flange and bolting. Refueling of the reactor is accomplished by removing the containment vessel closure head, installing a sealing collar between the CV and RV flanges (a permanent seal is provided between the CV and refueling cavity), and removing the RV head. The refueling cavity above the containment and RV is then flooded, and the RV internals are removed and stored in the refueling cavity. Fuel assemblies are vertically lifted from the RV directly into a fuel handling and storage area, using a refueling machine located above the CV. Thus, no refueling equipment is required inside containment, and the single refueling machine, located in the fuel handling area, is used for all fuel movement activities. In addition, this arrangement eliminates the in-containment polar crane, since all heavy reactor components are accessed through the containment closure head and are handled by the fuel handling over-head bridge crane. Spent fuel high density storage racks include integral neutron absorbing material to maintain the required degree of subcriticality. The racks include storage locations for 356 fuel assemblies, thus providing a minimum storage capability corresponding to 18 years of plant operation, with room for a full core off-load.

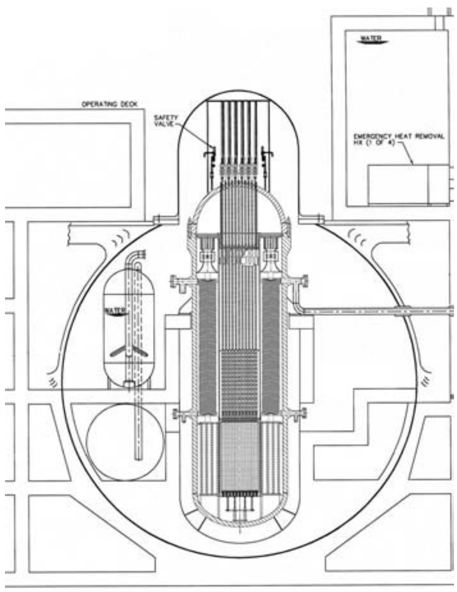


FIG. 5.7-4. IRIS Spherical Steel Containment

5.7.2.4 Primary components

Reactor pressure vessel

The IRIS reactor vessel (RV) is an integral configuration, which houses not only the nuclear fuel and control rods, but also all the major reactor coolant system (RCS) components (see Figure 5.7-1). This includes: eight small, spool type, reactor coolant pumps (RCPs); eight modular, helical coil, once through steam generators (SGs); a steel reflector which surrounds the core in the RV downcomer to improve neutron economy and reduce neutron fluence on the RV; and a pressurizer located in the RV upper head. This simplified integral arrangement eliminates the individual component pressure vessels and large connecting loop piping between them, resulting in a compact, more economic configuration and in the elimination of the large loss-of-coolant accident as a design basis event. Because the IRIS integral vessel contains all the RCS components, it is larger than a traditional RV, and has an ID of 6.2 meters and an overall height of 21.3 meters including the closure head.

The major in-vessel components are described below:

- **Reactor core** (previously described in Section 5.7.2.2)

- **Reactor coolant pumps (RCP)** - An advanced RCP has been adopted as the reference for the IRIS reactor. IRIS will feature a "spool type" pump that has been used in marine and chemical plant applications requiring high flow rates and low developed head. The motor and pump consist of two concentric cylinders, where the outer ring is the stationary stator and the inner ring is the rotor that carries high specific speed pump impellers. The spool type pump is located entirely within the reactor vessel; only small penetrations for the electrical power cables are required. High temperature motor windings and bearing materials are being developed to eliminate any need for cooling water and the associated piping penetrations through the RV. The spool pump geometric configuration provides high inertia/coastdown and high run-out flow capability that will contribute to mitigate the consequences of Loss-Of-Flow Accidents (LOFAs). Because of their low developed head, spool pumps have never been candidates for nuclear applications in loop reactors. However, the IRIS integral RV configuration and low coolant loop pressure drop can accommodate these pumps and take full advantage of their unique characteristics.

- **Steam generators** – The IRIS SGs employ a once-through, helical-coil tube bundle design with the primary fluid outside the tubes. Eight steam generator modules are located in the annular space between the core barrel (outside diameter 2.85 m) and the reactor vessel (inside diameter 6.2 m). Each IRIS SG module consists of the tubes, the lower feed water and upper steam headers, a central inner column to support them, and an outer wrapper. The enveloping outer diameter of the tube bundle is 1.64 m. Each SG has 656 tubes, and the tubes and headers are designed for the full external RCS pressure. The tubes are connected to the vertical sides of the lower feedwater header and the upper steam header. The SG is supported from the RV wall and the headers are bolted to the vessel from the inside of the feed inlet and steam outlet pipes. A double gasket, with a monitor leak-off, provides the pressure boundary between the primary coolant and the secondary side feed water inlet and steam outlet penetrations in the reactor vessel.

  Feedwater enters the SG through a nozzle in the reactor vessel wall and the lower header. As it flows upwards inside the SG tubes toward the upper steam header, it is heated to saturation temperature, boiled, and superheated. Steam then exits the SG through the nozzle in the reactor vessel wall.

  The helical SG tube bundle is contained within an outer wrapper (flow shroud) that directs the primary water flow from the top of the SG, downward through the bundle (outside the tubes), and out the bottom of the bundle into the reactor vessel downcomer region. Each of the eight reactor coolant pumps is attached directly to the top of its corresponding SG flow shroud, so that its flow is entirely directed through the SG bundle region. The helical-coil tube bundle design is capable of accommodating thermal expansion without excessive mechanical stress, and has high resistance to flow-induced vibrations. A prototype of this SG was successfully tested by IRIS team member Ansaldo Energia in an extensive test campaign conducted on a 20 MW(t) full diameter, part height, test article. The performance characteristics (thermal, vibration, pressure losses) were investigated along with the determination of the operating characteristics domain for stable operation.

- **Pressurizer** - The IRIS pressurizer is integrated into the upper head of the reactor vessel. The pressurizer region is defined by an insulated, inverted top-hat structure that divides the circulating reactor coolant flow path from the saturated pressurizer water. This structure includes a closed cell insulation to minimize the heat transfer between the hotter pressurizer fluid and the subcooled water in the primary water circulating flow path. Heater rods are located in the bottom portion of the inverted top-hat and are positioned outside the CRDM drive lines. The bottom portion of this inverted top-hat contains holes to allow water insurge and outsurge to/from the pressurizer region.

  By utilizing the upper head region of the reactor vessel, the IRIS pressurizer provides a very large water and steam volume, as compared to plants with a traditional, separate, pressurizer vessel. The IRIS pressurizer has a total volume of ~70 m^3, which includes a steam volume of ~50 m^3. This steam volume is about 1.6 times bigger than the AP1000 pressurizer steam space, while IRIS has ~1/3 the core power. This large steam volume to power ratio contributes to the fact that IRIS does not require the use of a pressurizer spray function to prevent the pressurizer safety valves from lifting for any design basis heatup transients.

5.7.2.5 Reactor auxiliary systems

Chemical and volume control system

The IRIS chemical and volume control system (CVCS) consists of a high pressure purification loop located inside containment, and the makeup and chemical addition portion of the system which is located outside containment. The inside containment, high design pressure portion of the CVCS includes the regenerative and letdown heat exchangers, demineralizers and filters, a canned circulation pump. and associated valves, piping, and instrumentation. The reactor coolant is demineralized, filtered, and returned to the reactor vessel without leaving the containment. The outside containment portion of the CVCS includes the makeup pumps, tanks, chemical and hydrogen addition equipment, and associated valves, piping, and instrumentation. The chemical and volume control system is designed to perform the following major tasks: Purification; Reactor coolant system inventory control and makeup; Chemical shim and chemical control; Oxygen control; Filling and pressure testing of the reactor coolant system; Borated makeup to auxiliary equipment; and, Pressurizer Auxiliary Spray.

Normal residual heat removal system

The normal residual heat removal system consists of two mechanical trains of equipment, each comprising one pump and one heat exchanger. Each train has a suction line from the reactor vessel and their discharge flow returns cooled water back to the reactor vessel via one of the two direct vessel injection connections. The normal residual heat removal system includes the piping, valves and instrumentation necessary for system operation. The major functions of this system are:

- Shutdown Heat Removal: The system removes both residual and sensible heat from the core and the reactor vessel. It reduces the temperature of the reactor coolant system during the second phase of plant cooldown. The first phase of cooldown is accomplished by transferring heat from the reactor coolant system via the steam generators to the main steam system. Following cooldown to 350°F with the steam generators, the normal residual heat removal system reduces the temperature of the reactor coolant system from 350° to 120°F (177 to 49 °C) within 96 hours after shutdown. The normal residual heat removal system then maintains the reactor coolant temperature at or below 120°F during the plant shutdown operations, until the plant is started up.

- Shutdown Purification: The system provides reactor coolant system flow to the chemical and volume control system during refueling operations.

- Low Temperature Overpressure Protection: The system includes safety relief valves that provide the low temperature overpressure protection function for the reactor coolant system during refueling, startup, and shutdown operations.

- Long-Term, Post-Accident Containment Inventory Makeup: The system provides a flow path for long-term post-accident makeup to the reactor vessel and containment, if and when required, to maintain inventory, under design assumptions of containment leakage.

Radioactive waste management

The IRIS reactor plant is designed to deal with liquid, gaseous and solid radioactive waste. The liquid waste systems include the radioactive waste drain system and the liquid radwaste system that collects and treats all water drained from the reactor. Treated liquid is stored and monitored before being discharged in a controlled manner. The gaseous radwaste system is a once-through, ambient-temperature, charcoal delay system. The solid waste management system is designed to collect and accumulate spent ion exchange resins, deep bed filtration media, spent filter cartridges, dry active wastes, and mixed wastes generated as a result of normal plant operation, including anticipated operational occurrences. The system is located in the auxiliary and radwaste buildings. Processing and packaging of wastes takes place in the radwaste building where the waste is stored until it is shipped offsite to a disposal facility.

5.7.2.6 Operating characteristics

The plant control scheme will be specifically designed for operation with the once-through steam generators and will be based on the "reactor follows plant load" strategy. A grid fluctuation will be automatically compensated for through turbine control valves in case of a frequency drop, and a decrease in pressure at the turbine will result in an increase in reactor power. IRIS is designed, similar to AP600/AP1000, to withstand the following operational occurrences without the generation of a reactor trip or actuation of the safety related passive engineered safety systems:

- ±5%/minute ramp load change within 15% and 100% power,
- ±10% step load change within 15% and 100% power,
- 100% generator load rejection,
- 100-50-100% power level daily load follow over 90% of the fuel cycle life,
- Grid frequency changes equivalent to 10% peak-to-peak power changes at 2%/minute rate,
- Loss of a single feedwater pump.

5.7.3 Description of the turbine generator plant systems

5.7.3.1 Turbine generator plant

The IRIS turbine design will consist of an 1800-rpm machine with a double-flow, high-pressure cylinder and one double-flow, low-pressure cylinder with exhausts to individual condensers. The turbine generator is intended for base load operation and also has load follow capability. The single direct-driven generator is gas-cooled and rated at 447 MVA at 22 kV, and a power factor of 0.9. Steam from the eight steam generators is combined into four steam line penetrations through the containment. These four lines extend to the high-pressure turbine through four stop valves and four governing control valves.

The turbine-generator and associated piping, valves, and controls are located completely within the turbine building. There are no safety-related systems or components located within the turbine building. The IRIS building design and layout will preclude the possibility of postulated turbine-generator high-energy missiles affecting safety-related structures, systems, or components so that the failure of the turbine-generator equipment does not preclude the safe shutdown of the reactor. The turbine-generator components and instrumentation associated with turbine-generator over-speed protection are accessible under operating conditions, so that the risks of over-speed events are minimized.

5.7.3.2 Condensate and feedwater systems

The IRIS condensate and feedwater systems supply the steam generators with clean, heated feedwater in a traditional, closed, steam cycle using regenerative feedwater heating. Full-flow cleanup of the condensate is provided to minimize deposits in the IRIS once-through steam generators. The feedwater cycle consists of seven stages of feedwater heating with two parallel string, low-pressure feedwater heaters located in the condenser neck with the next two single-string, low-pressure heaters, deaerator, and the high-pressure heaters located within the turbine building. The condenser hotwell and deaerator storage capacity provides sufficient storage to prevent minor, short-duration mismatches in flow from affecting plant operation. This margin, coupled with three 50 percent condensate and main feedwater pumps, provides operational flexibility and the ability for an operator to control feedwater and condensate transients.

5.7.4 Instrumentation and control systems

The I&C system design is integrated and will be based on the latest digital technology. Anticipated benefits from this technology will build on the already advanced I&C used in the AP600/1000 designs.

5.7.4.1 Design concept, including control room

The IRIS instrumentation and control architecture will be arranged in a hierarchical manner to provide a simplified, structured design that is horizontally and vertically integrated. Information is extracted from a data highway/monitor bus to control centers and data displays that facilitate the interaction between the plant operators and the I&C. The portions of the I&C that perform the protective, control, and data monitoring functions operate directly from the plant sensors. These include the protection and safety monitoring system, the plant control system, and the in-core instrumentation system. The plant control system has the function of establishing and maintaining the plant operating conditions within prescribed limits. The control system improves plant safety by minimizing the number of situations for which some protective response is initiated and relieves the operator from routine tasks. The protection and safety monitoring system is the safety grade system that provides reactor trip and safety component actuation signals. It is designed to be highly redundant and to prevent common mode failures. However, in the low-probability case where a common mode failure could occur, a diverse actuation system (DAS) provides an alternative means of initiating the reactor trip and emergency safety features. The hardware and software used to implement the DAS are different from the hardware and software used to implement the protection and safety monitoring system. The DAS has been included to meet the anticipated transient without (reactor) scram (ATWS) rule and to reduce the probability of a severe accident resulting from the unlikely coincidence of a transient and common mode failure of the protection and safety monitoring.

Control Room

The IRIS operation and control will be provided from an advanced main control room that incorporates the latest man-machine interface features and advanced display and control technologies. In addition, IRIS will include a separate remote shutdown workstation, a waste processing control room, and a technical support center. The main control room and the remote shutdown workstation are the signal interfaces with the plant components. These interfaces are via the plant protection and safety monitoring system processor and logic circuits, which interface with the reactor trip and engineered safety features plant components; the plant control system processor and logic circuits, which interface with the non-safety plant components; and the plant monitor bus, which provides plant parameters, plant component status, and alarms.

Sharing of a single, common control room by multiple IRIS modules is being investigated.

5.7.4.2 Reactor protection and other safety systems

The IRIS design includes instrumentation and controls to automatically sense transient or accident situation, trip the reactor, and initiate as required the engineered safety features with no need for operator actions. These actions are designed to prevent damage to the core, as well as mitigate the consequences of the postulated events and provide containment integrity. The protection and safety monitoring system provides the safety-related functions necessary during normal operation, to shut down the plant, and to maintain the plant in a safe shutdown condition. The protection and safety monitoring system controls the safety-related components in the plant that are operated from the main control room or remote shutdown workstation.

5.7.5 Electrical systems

The IRIS on-site power system design concept will be similar to the on-site power systems developed for the other Westinghouse advanced passive plants. The on-site power system is designed to provide reliable electric power to the plant safety and non-safety equipment for normal plant operation, startup, and normal shut down, and for accident mitigation and emergency shutdown. The on-site power systems include the main AC power system and the DC power system.

5.7.5.1 Operational power supply systems

The main AC power system is a non-Class IE system that does not perform any safety function. The standby power supply is included in the on-site standby power system. The power to the main AC power system normally comes from the station main generator through unit auxiliary transformers. The plant is designed to sustain a load rejection from 100 percent power with the turbine generator continuing stable operation while supplying the plant house loads. The on-site standby AC power system is powered by the two on-site standby diesel generators and supplies power to selected loads in the event of loss of the normal AC power supplies.

The plant DC power system comprises two independent systems, one Class IE and one non-Class IE DC power systems. Each system consists of ungrounded stationary batteries, DC distribution equipment, and uninterruptible power supplies.

5.7.6 Safety Concept

5.7.6.1 Safety requirements and design philosophy

IRIS has been primarily focused on featuring a design with innovative safety characteristics. The IRIS design provides for multiple levels of defense for accident prevention and mitigation (defense-in-depth), resulting in extremely low core damage probabilities while minimizing the occurrences of containment flooding, pressurization, and heat-up situations.

The very first line of defense in the IRIS defense in depth approach is to eliminate initiators that could convincibly lead to core damage. This concept is implemented through the "safety by design" approach, which can be simply described as "design the plant in such a way to eliminate the accidents from occurring, rather than coping with their consequences". If it is not possible to eliminate the accidents altogether, then the design should be such to inherently reduce their consequences and/or decrease their probability of occurring. The key difference from previous practice is that the integral reactor design is intrinsically conducive to eliminating accidents, to a degree impossible in conventional loop-type reactors. The elimination of the large LOCAs, since no large primary penetrations of the reactor vessel or large loop piping exist, is only the most easily visible of the safety potential characteristics of integral reactors. Many others are possible, but they must be carefully exploited through an appropriate design that is kept focused on selecting design characteristics that are most amenable to eliminating accident initiators. IRIS has strived to achieve that and some of the main results are summarized in Table 5.7-II, which illustrates the implications of the safety by design approach, and in Table 5.7-III, that describes the effect of safety by design on some typical design basis events for LWRs.

TABLE 5.7-II. IMPLICATIONS OF SAFETY BY DESIGN APPROACH

IRIS Design Characteristic	Safety Implication	Accidents Affected
Integral Layout	No large primary piping	- LOCAs
Large, Tall Vessel	Increased water inventory	- LOCAs - Decrease in heat removal
	Increased natural circulation	- Various events
	Can accommodate internal CRDMs	- RCCA ejection - Eliminate head penetrations
Heat Removal from inside the vessel	Depressurizes primary system by condensation and not by loss of mass	- LOCAs
	Effective heat removal by SG/EHRS	- LOCAs - All events for which effective cooldown is required - ATWS
Reduced size, higher design pressure containment	Reduced driving force through primary opening	- LOCAs
Multiple coolant Pumps	Decreased importance of single pump failure	- Locked rotor, shaft seizure/break
High design pressure steam generator system	No SG safety valves	
	Primary system cannot over-pressure secondary system	- Steam generator tube rupture
	Feed/Steam System Piping designed for full RCS pressure reduces piping failure probability	- Steam line break - Feed line break
Once Through steam generator	Limited water inventory	- Steam line break - {Feed line break}
Integral Pressurizer	Large pressurizer volume/reactor power	- Overheating events, including feed line break. - ATWS

TABLE 5.7-III. IRIS RESPONSE TO PWR CLASS IV EVENTS

Design Basis Condition IV Events		Effect of IRIS Safety-by-Design
1	Large Break LOCA	- Eliminated by design (no large piping)
2	Steam Generator Tube Rupture	- Reduced consequences, simplified mitigation
	Steam System Piping Failure	- Reduced probability, reduced (limited containment effect, limited cooldown) or eliminated (no potential for return to critical power) consequences
4	Feedwater System Pipe Break	- Reduced probability, reduced consequences (no high pressure relief from reactor coolant system)
5	Reactor Coolant Pump Shaft Break	- Reduced consequences
6	Reactor Coolant Pump Shaft Seizure	
7	Spectrum of RCCA ejection accidents	- Potentially eliminated by design, requires development of internal CRDM
8	Design Basis Fuel Handling Accidents	- No impact

The IRIS defense-in-depth capability next includes the traditional multiple levels of defense for a very wide range of plant events, similar to AP600/AP1000. Defense-in-depth is built into the IRIS design, where the design goal is to always maintain the core covered with water and avoid fuel damage, with a multitude of individual plant features capable of providing some degree of defense of plant safety. In addition to the safety by design, the following five additional aspects of the IRIS design contribute to defense-in-depth.

Stable Operation. In normal operation, the most fundamental level of defense-in-depth ensures that the plant can be operated stably and reliably. This is achieved by the selection of materials, by quality assurance during design and construction, by well-trained operators, and by an advanced control system and plant design that provide substantial margins for plant operation before approaching safety limits.

Physical Plant Boundaries. One of the most recognizable aspects of defense-in-depth is the protection of public safety through the physical plant boundaries. Releases of radiation are directly prevented by the fuel cladding, the reactor pressure boundary, and the containment pressure boundary. For the fuel cladding boundary, the reactor protection system is designed to actuate a reactor trip whenever necessary to prevent exceeding the fuel design limits. The core design, together with defense-in-depth process and decay heat removal systems, provides this capability under expected conditions of normal operation, with appropriate margin for uncertainties and anticipated transient situations. The reactor coolant pressure boundary is designed with complete overpressure protection and appropriate materials to provide and maintain the boundary during all modes of plant operation. The containment vessel, in conjunction with the defense-in-depth heat removal systems, is designed so that: its design pressure is not exceeded following postulated design basis accidents; a large margin to the design basis pressure is maintained during postulated design basis accidents to minimize leakage probability; and, containment failure does not occur even under severe accident conditions.

Passive Safety-Related Systems. The safety-related passive systems are sufficient to automatically establish and maintain core cooling and containment integrity for the plant following design basis events, assuming that the most limiting single failure occurs. The safety-related passive systems use only natural forces, such as gravity and natural circulation for their continued operation. No pumps, fans, diesels, chillers, or other rotating machinery are used. A few simple valves align the passive safety systems when they are automatically actuated by the safety-related protection and monitoring system (PMS). To provide high reliability, these valves are designed to actuate to their safety positions upon loss of power or upon receipt of a safety actuation signal. However, they are also supported by multiple, reliable power sources to avoid unnecessary actuations. The passive systems are designed to meet the single-failure criteria, and probabilistic risk assessments (PRAs) are used to verify their reliability. The PMS provides the safety-related functions of reactor trip, engineered safety features actuation, and post-accident monitoring. The IRIS design basis for the PMS is to provide an automatic response to any postulated accident, without requiring any operator action for extended periods of time (more than 3 days).

Non-safety Systems. The next design level of defense-in-depth is the availability of certain non-safety systems for reducing the potential for events leading to core damage. For more probable events, these defense-in-depth, non-safety systems automatically actuate to provide a first level of defense to reduce the likelihood of unnecessary actuation and operation of the safety-related systems. These non-safety-related systems establish and maintain safe shutdown conditions for the plant following design basis events, provided that at least one of the non-safety-related AC power sources is available.

Also, to minimize core damage probability, diverse, non-safety systems are provided to back up the main functions of the passive safety related systems. These systems are being defined on the basis of PRA considerations so to minimize the core damage and the radioactivity release probabilities. This diversity exists, for example, in the residual heat removal function. The emergency heat removal system (EHRS) is the passive safety-related feature for removing decay heat during a transient. In case of multiple failures in the EHRS, defense-in-depth is provided by a simple, non-safety, passive containment cooling system and by the gravity driven injection from the pressure suppression system tanks and automatic depressurization (passive feed and bleed) functions. The introduction of these diverse features in the design is made amenable by the intrinsic characteristics of the integral layout, as exploited in the safety by design approach.

In addition to diversity, redundancy is also a traditional requirement for defense-in-depth. As it is common procedure, IRIS has extensive redundancy for critical components such as pumps and steam generators. In fact, due to adoption of multiple pumps, a typical class IV accident such as the pump shaft break/seizure in IRIS will not result in core damage even without reactor trip.

Containing Core Damage. IRIS is designed so that the reactor cavity is flooded following any severe accident event that may have the potential for core uncovery and melting. The objective of this cavity flooding action is to prevent reactor vessel failure and subsequent relocation of molten core debris into the containment. Retention of the debris in the vessel significantly reduces the uncertainty in the assessment of containment failure and radioactive release to the environment due to ex-vessel severe accident phenomena. It must be emphasized that IRIS is designed to avoid core uncovery and consequently melting, under all accident conditions. The capability of in-vessel core retention is an added feature.

5.7.6.2 Safety systems and features (active, passive and inherent)

The safety by design approach, together with the use of passive safety systems provides improvements in plant simplification, safety, reliability, and investment protection over conventional plant designs. The IRIS passive systems design complements the safety by design approach and the consequent elimination of some postulated design basis events (large LOCAs) and the inherent mitigation of several other (steam generator tube rupture, steam line break, locked rotor,…) through the definition of a safety strategy that is specifically tailored to respond to those remaining accident initiators that are main contributors to core damage frequencies. This design approach allows the licensing safety criteria to be satisfied with a greatly simplified plant design. The IRIS passive safety systems are simpler than previous passive safety designs since they contain significantly fewer components, reducing the required tests, inspections, and maintenance, require no active support systems, and their readiness is easily monitored.

Passive Core and Containment Cooling

The IRIS passive systems configuration is presented in Figure 5.7-5, and includes:

- A passive emergency heat removal system (EHRS) made of four independent trains, each including a horizontal, U-tube heat exchanger located in the Refueling Water Storage Tank (RWST) located outside the containment structure that is connected to a separate SG feed/steam line. The RWST provides the heat sink for the EHRS heat exchangers. The EHRS is sized so that a single train can provide decay heat removal in the case of a loss of secondary system heat removal capability. The EHRS operates by natural circulation removing heat from the primary system through the steam generators heat transfer surface, condensing the steam produced in the EHRS heat exchanger, and transferring the heat to the RWST, and returning the condensate back to the SG. The EHRS provides the main post-LOCA depressurization (depressurization without loss of mass) and coolant makeup function for IRIS because it condenses the steam produced by the core directly inside the reactor vessel thus minimizing the break flow, while transferring the decay heat to the environment, thus performing the functions of both core cooling and containment depressurization;

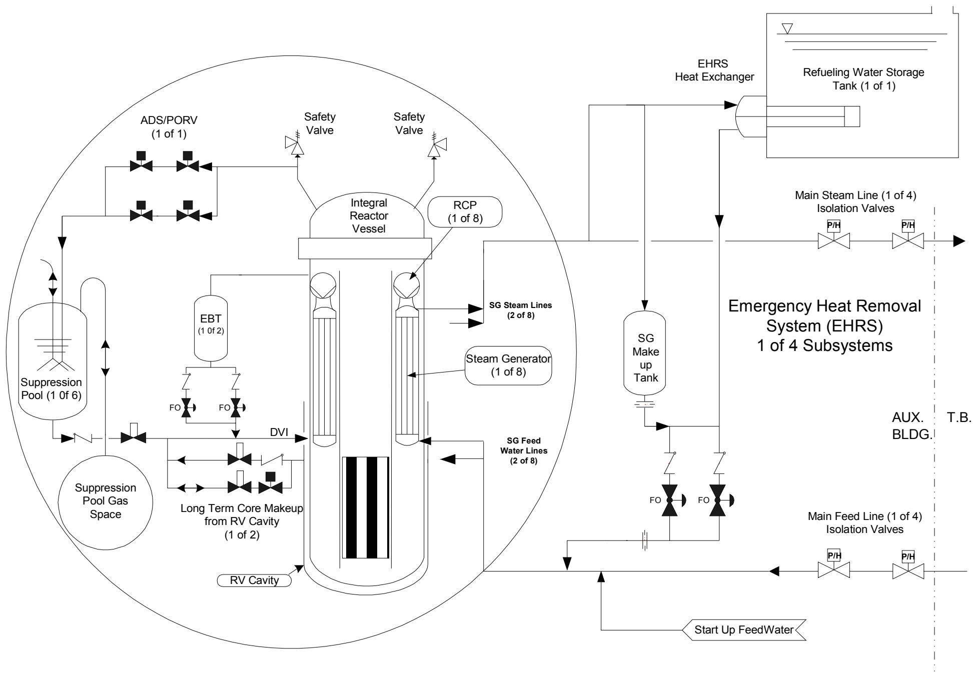


FIG. 5.7-5. Passive core and containment cooling system.

- Two compact (450 ft^3) full-system pressure emergency boration tanks (EBTs) which deliver emergency boration through the direct vessel injection (DVI) lines for transient events. By their operation these tanks also provide a limited gravity feed makeup water to the primary system;

- A small automatic depressurization system (ADS) connected to the pressurizer steam volume, which assists the EHRS in depressurizing the reactor vessel when/if the reactor vessel coolant inventory drops below a specific setpoint. This ADS has one stage and consist of two parallel 4 inch lines, with two normally closed valves. The single ADS line downstream of the closed valves discharges into the pressure suppression system pool tanks through a sparger. This ADS function ensures that the reactor vessel and containment pressures are equalized in a timely manner limiting the loss of coolant and thus preventing core uncovery following postulated LOCAs;

- A containment Pressure Suppression System (PSS) which consists of 6 water tanks and a common tank for non-condensable gas storage. Each suppression water tank is connected to the containment atmosphere through a vent pipe connected to a submerged sparger to condense steam released in the containment following a loss of coolant or steam/feed line break accident. The suppression system limits the peak containment pressure following a blowdown event to less than the containment design pressure. The suppression system water tanks also provide an elevated source of water that is available for gravity injection into the reactor vessel through the DVI lines in the event of a loss of coolant accident (LOCA);

- A specially constructed lower containment volume that collects the liquid break flow, as well as any condensate from the containment in a cavity where the reactor vessel is located. During a LOCA, the cavity floods above the core level, creating a gravity head of water sufficient to provide coolant makeup to the reactor vessel through the DVI lines.

The safety strategy of IRIS provides a diverse means of core shutdown by makeup of borated water from the EBT and core cooling and heat removal to the environment through the EHRS in the event that normally available active systems are not available. In the event of a significant loss of primary-side water inventory, the primary line of defense for IRIS is the large coolant inventory in the reactor vessel and the fact that depressurization is attained with very limited loss of mass, thus maintaining a sufficient inventory in the primary system and guaranteeing that the core will remain covered for all postulated events. The EBT is capable of providing some primary system injection at high pressure, but this is not necessary since the IRIS strategy relies on "maintaining" coolant inventory, rather than "injecting" makeup water. This strategy ensures that the core remains covered with water for an extended period of time (days and possibly weeks). Of course, when the reactor vessel is depressurized to near containment pressure, gravity flow from the suppression system and from the reactor will maintain the coolant inventory for an unlimited period of time.

Main control room habitability system

The main control room habitability system (VES) provides fresh air, cooling, and pressurization to the main control room (MCR) following a plant accident. Operation of the VES is automatically initiated upon receipt of a high MCR radiation signal, which isolates the normal control room ventilation path and initiates pressurization. Following system actuation, all functions are completely passive. The VES air supply is contained in a set of compressed air storage tanks. The VES also maintains the MCR at a slightly positive pressure, to minimize the infiltration of airborne contaminants from the surrounding areas.

Containment isolation

IRIS containment isolation further builds on the AP600/AP1000 design philosophy which significantly improved isolation over that of conventional PWRs. Major improvements are made in reducing the number and size of penetrations. Furthermore, the number of normally open penetrations is significantly reduced. There are no containment penetrations required to support post-accident mitigation functions.

Long-term accident mitigation

A major safety advantage of IRIS versus current PWRs is that long-term accident mitigation is maintained without operator action and without reliance on offsite or onsite AC power sources. The passive safety systems are designed to provide long-term core cooling and decay heat removal without the need for operator actions and without reliance on the active non-safety-related systems for 7 days.

Deterministic Design Basis

The safety by design approach with its vastly enhanced defense in depth provides an effective method of satisfying regulatory requirements for design basis events. The main effects of this approach on IRIS safety were listed in Table 5.7-II and 5.7-III and are discussed here in some detail:

- **Loss of Coolant Accidents – LOCAs.** The integral layout eliminates by design the possibility of large break LOCAs, since no large primary system piping is present in the reactor coolant system. Also, the probability of small break LOCA is lessened because of the drastic reduction in overall piping length, and the largest primary piping is limited to a diameter of less than 4 in. To cope with postulated small break LOCA, an innovative strategy has been developed to fully exploit the IRIS design characteristics, such as:

1) The initial large coolant inventory in the reactor vessel;
2) The EHRS which removes heat directly from inside the RV thus depressurizing the RV by condensing steam, rather than depressurizing by discharging mass;
3) The compact, small diameter, high design pressure containment that assists in limiting the blowdown from the RV by providing a higher back-pressure in the initial stages of the accident and then rapidly equalizing the vessel and containment pressures.

As illustrated in Figure 5.7-6, IRIS is designed to limit the loss of coolant from the vessel rather than relying on active or passive systems to inject water into the RV.

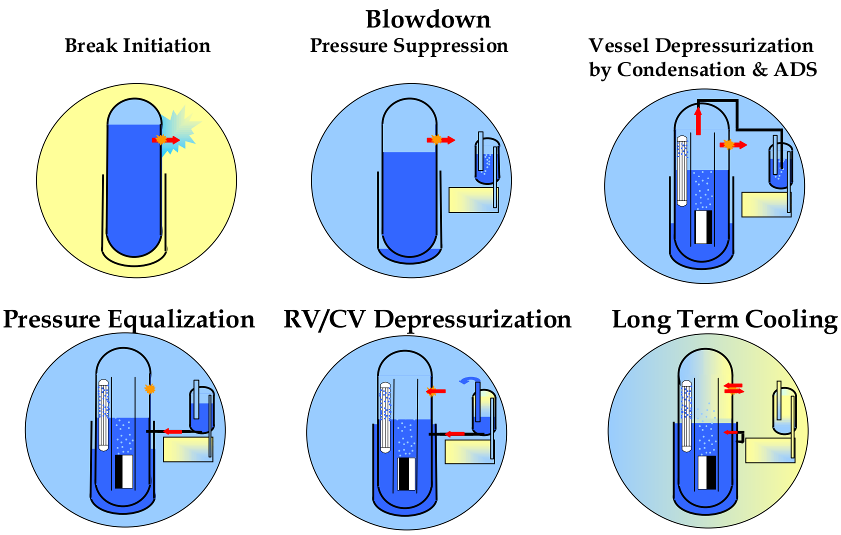


FIG. 5.7-6. Overview of IRIS response to loss of coolant accidents.

After the LOCA initiation, the RV depressurizes and loses mass to the containment vessel (CV) causing the CV pressure to rise (Blowdown phase). The mitigation sequence is initiated with the reactor trip and pump trip; the EBTs are actuated to provide boration; the EHRS is actuated to depressurize the primary system by condensing steam on the steam generators (depressurization without loss of mass); and finally the ADS is actuated to assist the EHRS in depressurizing the RV. The containment pressure is limited by the Pressure Suppression System and the reduced break flow due to the EHRS. At the end of the blowdown phase the RV and CV pressures become equal (pressure equalization) with a CV pressure peak <8 bar$_g$. The break flow stops and the gravity makeup of borated water from suppression pool becomes available. The coupled RV/CV system is then depressurized (RV/CV depressurization phase) by the EHRS (steam condensation inside the RV exceeds decay heat boiloff). In this phase the break flow reverses since heat is removed not from the containment, but directly inside the vessel. As steam from the containment is condensed inside the pressure vessel, the containment pressure is reduced, and a portion of suppression pool water is pushed out through the vents and assists in flooding the vessel cavity.

The depressurization phase is followed by the long term cooling phase (RV and CV pressure reduced to <2 bar_g in <12 hours), during which the gravity makeup of borated water from both suppression pool and RV cavity are available as required. Since decay heat is directly removed from within the vessel and the vessel and containment are thermo-hydraulically coupled, the long term break flow does not correspond to the core decay heat, but in fact it is limited to only the containment heat loss.

- **Steam Generator Tube Rupture.** In IRIS, the steam generator tubes are in compression (the higher pressure primary fluid is outside the tubes) and the steam generators headers and tubes are designed for full external reactor pressure. Thus, tube rupture is much less probable and if it does occur, tube failure propagation is practically impossible. Moreover, since the steam generators, the feed and steam piping and the isolation valves are designed for full reactor coolant system pressure, a tube rupture event is rapidly terminated by closure of the main steam and feed isolation valves upon detection of the failure. Once the isolation valves are closed, no release of radioactivity (primary fluid) will be possible, and the primary water will simply fill the faulted steam generator. Given the limited volume of the steam generators and piping, no makeup to the RV will be necessary. Compared to loop PWRs, no steam generator overfill-overpressure-water relief/safety valve failure, resulting in unisolable containment bypass scenario, is possible in IRIS. Also, the number of tubes assumed to fail has a limited effect on the system response and does not impact the final plant state.

- **Increase in Heat Removal from the Primary Side.** The limited water inventory in the once through steam generator has an important effect on the events in this category. Increases in heat removal due to increased steam flow are eliminated since the steam flow from the once-through steam generators cannot exceed feed water flow rate. Also, the consequence of a design basis steam line break events are lessened. Not only is the impact on the containment limited by the reduced discharge of mass/energy, but also no return to power due to the cooldown of the primary system is possible.

- **Decrease in Heat Removal from the Secondary Side.** Events in this category (loss of offsite power, loss of normal feedwater, turbine trip, feed system piping failure,…) could have larger consequences in IRIS than in loop type PWRs because of the limited water inventory in the once through steam generators. However, this is more than balanced by the large thermal inertia in the primary system (IRIS water inventory on a coolant-per-MW(t) basis is more than 5 times larger than in advanced passive PWRs), and by the large steam volume in the IRIS pressurizer (steam volume-to-power ratio is also more than 5 times that of the AP1000). The reactor trip setpoint is rapidly reached on a low feedwater signal, and the EHRS connected to the steam generators effectively removes sufficient heat to prevent any pressurizer overfill or high pressure relief from the reactor vessel to the containment.

- **Decrease in Reactor Coolant Flow Rate.** The IRIS response to a complete loss of coolant flow is comparable to that of the AP600/AP1000. For the design basis Locked Rotor event, IRIS response is improved over other PWRs by the increased number of pumps, which reduces the relative importance of a loss of a single pump flow. This design choice allows IRIS to prevent fuel damage (i.e. no departure from nucleate boiling) following a postulated locked rotor event even without a reactor trip.

- **Spectrum of Postulated Rod Ejection Accidents.** The integral reactor vessel provides a large space above the core that can be utilized to place the control rod drive mechanisms (CRDMs) inside the vessel. This in-vessel CRDM location would eliminate the rod ejection accident by design. Additionally, the operational failures associated with the vessel head penetrations would be eliminated since there are no large drive line penetrations. However, the internal CRDMs have still not been proven for large integral reactors of the IRIS size, and their state of development is incompatible with the current IRIS schedule. Thus, the reference IRIS design features a traditional drive mechanism. The development of in vessel CRDMs is actively being pursued and the option is left open to modify the reference design if warranted by technical developments.

- **Increase in reactor coolant inventory.** This category of events is all but eliminated in IRIS since IRIS does not utilize high-pressure coolant injection following a LOCA.

5.7.6.3 Severe accidents (beyond design basis accidents)

In-vessel retention of molten core debris

The IRIS is designed to provide in-vessel retention of core debris by depressurizing and cooling the outside of the reactor vessel following severe accidents. IRIS has reactor vessel insulation that promotes in-vessel retention and surface treatment that promotes wetability of the external surface.

The design features of the containment ensure flooding of the vessel cavity region during accidents and submerging the reactor vessel lower head in water. Liquid effluent released through the break during a LOCA event is directed to the reactor cavity. The IRIS design also includes a provision for draining part of the pressure suppression system (PSS) water tanks water into the reactor cavity.

The IRIS design also includes a second means of containment cooling should cooling via the EHRS be defeated. In this event direct cooling of the containment outer surface is provided and containment pressurization is limited to less than its design pressure. This cooling plus multiple means of providing gravity driven makeup to the core provides a diverse means of preventing core damage and ensuring containment integrity and heat removal to the environment.

5.7.7 Plant layout

5.7.7.1 Buildings and structures, including plot plan

Two plant arrangements of multiple IRIS reactor single units and twin units have been considered to establish the IRIS site plot plans. Further optimization will occur to increase the shared facilities and systems not only within a twin unit but also among single reactor units and twin units with the aim of reducing the plant overall footprint. The data provided here should be considered as preliminary enveloping data.

- Independent Multiple Single Unit Arrangement – The three single unit arrangement (Figure 5.7-7) shows three IRIS modules that are completely independent, including their own non-safety related service water and main circulating water mechanical draft cooling towers. This arrangement is based on the assumption that the units would be constructed in series in a “slide-along” manner. The units would be started up in sequence as construction, pre-operation testing, fuel load, and startup testing are all completed for a unit. The units are spaced sufficiently apart so that the first completed unit could be operated while construction of the subsequent unit(s) is still in progress, by establishing a temporary exclusion zone between the operating unit(s) and the unit(s) under construction. This arrangement and construction sequencing is aimed at minimizing the construction time of a unit and at providing the utility with generating capability as soon as possible. Other advantages of this slide-along construction method are envisioned to be shorter construction time for the subsequent units by taking advantage of the experience of the work force.

- Multiple Twin-Unit Arrangement – The two twin-unit arrangement (Figure 5.7-8) shows two independent, twin unit reactors. This arrangement is aimed at maximizing shared components between the two reactors comprising one twin-unit, yet maintaining the ability to initiate operation of a completed twin-unit while construction of subsequent twin(s) proceeds in a "slide-along" manner. Each twin-unit is completely independent from the subsequent twin(s) and each reactor within a twin has its own turbine generator, condenser, and feed and steam systems, contained in a single T/G building with their own non-safety service water and main circulating water mechanical draft cooling towers. However, within a twin-unit, many systems, functions, and physical facilities are shared including: (back to back) control rooms, fuel handling area with refueling machine and spent fuel pit and cask loading facility, radwaste treatment, support systems, and switchyard. Within the twin-unit, separate safety grade power supplies, protection cabinets and switchgear, and electrical systems are maintained.

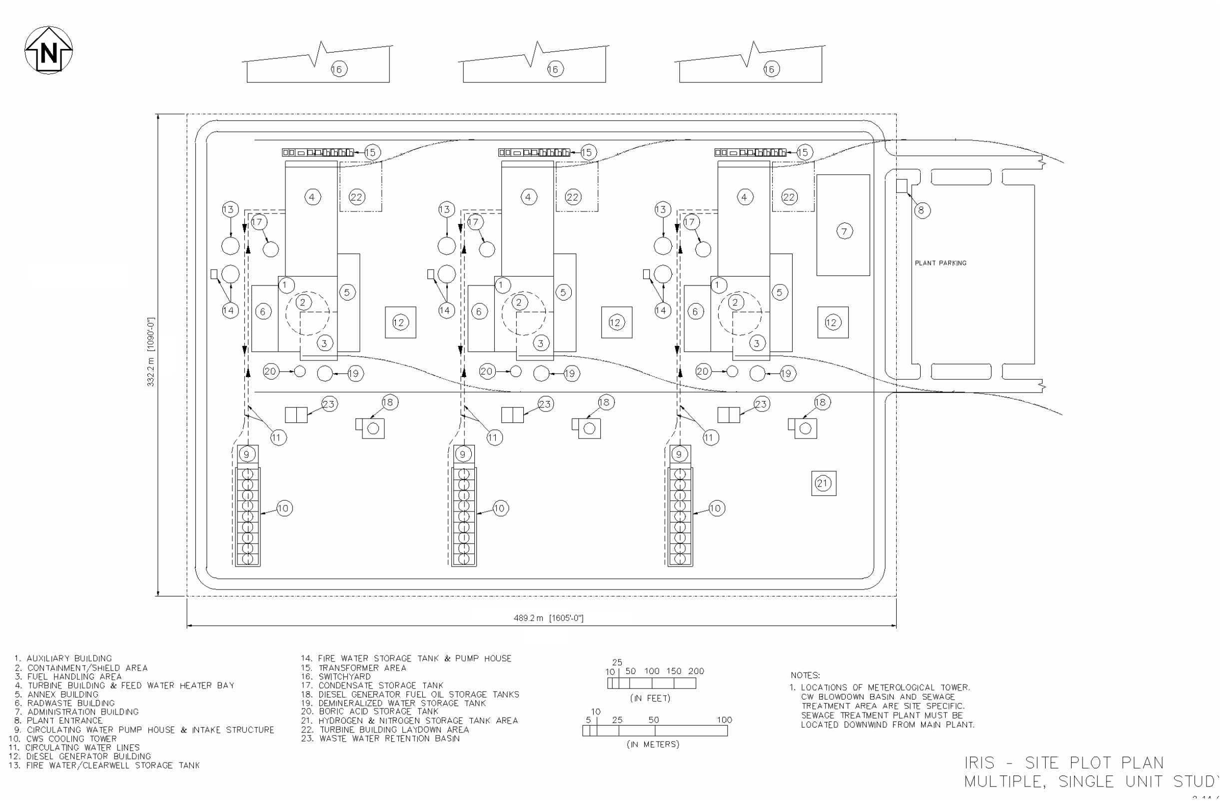


FIG. 5.7-7. IRIS, Three Single Unit Site Plot Plan.

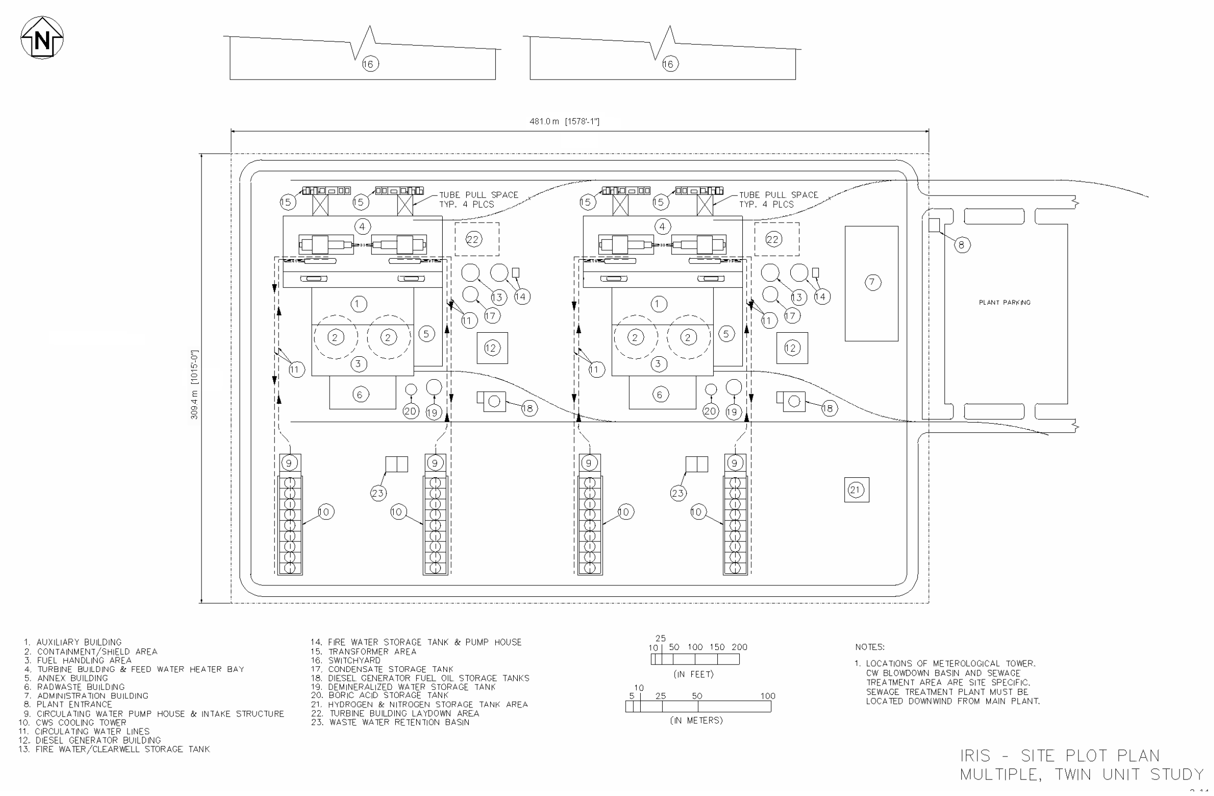


FIG. 5.7-8. IRIS, Two Twin-Unit Site Plot Plant.

The current preliminary plant arrangement for the site plan (Figures 5.7-7 and 5.7-8) has established the bottom of the basemat of the seismic block at –15 meters. The plant grade level is at 0 meters. This seismic basemat includes the containment and shield structure (Item 2), surrounded by the auxiliary building, which includes the control room and all safety related equipment and fuel handling equipment (Item 1). The roof elevation of the auxiliary building, which surrounds and covers the containment and shield building is at +32 meters above grade. The fuel handling area (Item 3) occupies the southern portion of the auxiliary building and extends over the containment such that the containment and RV closure heads can be lifted vertically and stored in the fuel handling area during refueling operations.

5.7.8 Technical data

General plant data

Power plant output, gross		MW(e)
Power plant output, net	335	MW(e)
Reactor thermal output {core power 1000 MW(t)]	1002	MW(t)
Power plant efficiency, net		%
Cooling water temperature		°C

Nuclear steam supply system

Number of coolant loops	Integral RCS	
Primary circuit volume, including pressurizer	455	m^3
Steam flow rate at nominal conditions	503	kg/s
Feedwater flow rate at nominal conditions	503	kg/s
Steam temperature/pressure	317/5.8	°C/MPa
Feedwater temperature/pressure	224/6.4	°C/MPa

Reactor coolant system

Primary coolant flow rate	4700	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at core inlet	292	°C
Coolant outlet temperature, at riser outlet	328.4	°C
Mean temperature rise across core	38	°C

Reactor core

Active core height	4.267	m
Equivalent core diameter	2.413	m
Heat transfer surface in the core	2992	m^2
Fuel inventory	48.5	t U
Average linear heat rate	9.97	kW/m
Average fuel power density	20.89	kW/kgU
Average core power density (volumetric)	51.26	kW/l
Thermal heat flux, Fq	2.60	
Enthalpy rise, F_H	1.65	
Fuel material	Sintered UO_2	
Fuel assembly total length	5207	mm
Rod arrays	square, 17x17	
Number of fuel assemblies	89	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	25	
Number of structural spacer grids	10	
Number of intermediate flow mixing grids	4	
Enrichment (range) of first core	2.6-4.95	Wt % U-235
Enrichment of reload fuel at equilibrium core	≤ 5.0	Wt %U-235
Operating cycle length (fuel cycle length)	30-48	months
Average discharge burnup of fuel (nominal)	40000-65000	MWd/t
Cladding tube material	ZIRLO™	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly		kg
Active length of fuel rods	4267	mm
Burnable absorber, strategy/material	IFBA and Er	
Number of control rods	37	
Absorber rods per control assembly	24	
Absorber material	Ag-In-Cd (black), Ag-In-Cd/304SS (gray)	
Drive mechanism	Magnetic jack	
Positioning rate [in steps/min or mm/s]	45 steps/min	
Soluble neutron absorber	Boric acid	

Reactor pressure vessel

Cylindrical shell inner diameter	6210 mm	
Wall thickness of cylindrical shell	285 mm	
Total height	21300 mm	
Base material: cylindrical shell	Carbon steel	
RPV head	Carbon steel	
Liner	Stainless steel	
Design pressure/temperature	17.2/360	MPa/°C
Transport weight (lower part), and	1045	t

RPV head	167	t

Steam generators

Type	once through, vertical, helical coil	
Number [Thermal capacity 125 MW(t)]	8	
Heat transfer surface	1150	m^2
Number of heat exchanger tubes	656	
Tube dimensions	17.5/13.2	mm
Shroud outer diameter	1640	mm
Total height	8500	mm
Transport weight	35	t
Shroud and tube sheet material	Stainless steel	
Tube material	Inconel 690-TT	

Reactor coolant pump

Type	Spool Type	
Number	8	
Design pressure/temperature	17.2 /343.3	MPa/°C
Design flow rate (at operating conditions)	587.5	kg/s
Pump head	19.8	m
Power demand at coupling, cold/hot	225 kW	
Pump casing material	N.A.	
Pump speed	1800 rpm	

Pressurizer

Total volume	71.41	m^3
Steam volume: full power/zero power	48.96	m^3
Design pressure/temperature	17.2/360	MPa/°C
Heating power of the heater rods	2400	kW
Number of heater rods	90	
Inner diameter	(RPV Head)	
Total height	(RPV Head)	
Material	(RPV Head)	
Transport weight	(RPV Head)	

Pressurizer relief tank

Not applicable

Primary containment

Type	Pressure Suppression, steel	
Overall form (spherical/cyl.)	spherical	
Dimensions (diameter/height)	25/32	m
Free volume	4540	m^3
Design pressure/temperature (DBEs)	1300/200	kPa/°C
(severe accident situations)	1300/200	kPa/°C
Design leakage rate	0.1	%vol/day
Is secondary containment provided?	Missile Protection and release filtration provided	

Reactor auxiliary systems

Power supply systems

Turbine plant

Generator

Condenser

Condensate pumps

Condensate clean-up system

Feedwater tank

Feedwater pumps

Condensate and feedwater heaters

- no information provided

5.7.9 Measures to enhance economy and maintainability

Various features have been incorporated to *simplify the design*, *minimize construction time*, and *reduce total cost* by eliminating components and reducing bulk quantities and building volumes. Some of these features include:

- Integral layout results in a small, 25 meter diameter, containment vessel with no in-containment refueling. The small containment in turns results in a small plant footprint;
- The flat, common basemat design of the nuclear island minimizes construction cost and schedule;
- Integrated protection system, the advanced control room, distributed logic cabinets, multiplexing, and fiber optics, significantly reduce the quantity of cables, cable trays, and conduits;
- Stacked arrangement of the Class IE battery rooms, the DC switchgear rooms, the integrated protection system rooms, and the main control room eliminates the need for the upper and lower cable spreading rooms that are required in the current generation of PWR plants;
- Application of the passive safety systems replaces and/or eliminates many of the conventional mechanical safety systems that are typically located in the Seismic Category I buildings in the current generation of PWR plants;
- Short lead time (four years from owner's commitment to commercial operation) and construction schedule (two years) reduce financing cost;
- Modular construction and economy of many identical units (e.g. 32 small steam generators and pumps are used per dual twin unit of 1,400 MW(e)) reduce the construction and component cost.

The integral configuration has allowed IRIS to implement the safety-by-design approach. Importantly, xploitation of the integral design features, combined with the safety-by-design approach, greatly improves the reactor response to accident events. Consequently, the IRIS design has achieved significant *design simplifications*, by simplifying or eliminating components and systems (e.g., elimination of the high pressure core safety injection system, pressurizer spray, and steam generator relief valves, smaller ADS, etc.) and by reduction of other safety systems, which result in both *improved safety and economics*.

Another characteristic of IRIS is the *improved operational flexibility*. IRIS is capable of operating with long life, straight burn cores. Even though the reference design features a two-batch three-year fuel cycle, selected on the basis of ease of licensing and U.S. utilities preference, IRIS is capable of operating in straight burn with a core lifetime of about 8 years. However, the significant advantages connected with a long refueling period in reducing operation and maintenance (O&M) costs are lost if the reactor still has to be shut down on a 18 to 24 months cycle for routine maintenance and inspection. Thus, first and foremost, the IRIS primary system components are designed to have very high reliability to decrease the incidence of equipment failures in order to reduce the frequency of required inspections or repairs. Next, IRIS has been designed to extend the need for scheduled maintenance outages to at least 48 months. The basis of the design has been a study performed in the mid to late 90s by MIT for a PWR power station to identify required actions for extending the maintenance period from 18 to 48 months. Thus, IRIS provides a range of operating options, and is also designed *recognizing the URD and EUR utility requirements*. However, the uninterrupted operation for 48 months requires reliable advanced diagnostics. The IRIS project is currently investigating various technologies, either already proven or in advanced phase of development, to monitor the behavior of the in-core components.

Finally, IRIS is addressing licensing by relying on the excellent defense in depth provided by the safety-by-design, and also by adopting risk informed regulation based on PRA analyses to achieve further goals including demonstrating that IRIS does not need emergency response planning. This, besides being very attractive to power producers considering IRIS, will have a very strong impact on the public acceptance. Elimination of the evacuation zone would also mean very *significant further reduction of the total cost*.

Various features contribute to *reducing the O&M cost*, and *consequently the power generation cost*. They include:

- Longer cycle (up to 48 months) and less frequent maintenance, resulting in:
  - increased availability and capacity factors (IRIS is expected to comfortably satisfy and exceed the 95% target);
  - reduced personnel requirement and cost;
  - reduced maintenance manpower cost;
  - reduced personnel exposure;
  - reduced replacement energy cost.

- Extra shielding provided by the integral layout vessel ensures:
  - longer RV life (indefinite as far as radiation damage is concerned);
  - significantly reduced activation of the RV and ex-RV structures;
  - reduced exposure during maintenance and D&D activities;
  - practically cold vessel, with potential to dispose it as non radioactive material, thus significantly reducing the D&D cost, and enabling simpler return to green field;
  - due to very low fast neutron fluence, no need for RV surveillance program, which further reduces cost.

- Improved fuel utilization and *reduced fuel cycle cost* is achieved through:
  - enhanced moderation;
  - use of radial reflector;
  - use of enriched boron reduces the volume (and cost) of chemical waste to process;
  - additionally, the fuel design is amenable to high burnup (robust design, increased plenum volume) and thus to extending the cycle length beyond 4 years.

As a result of all these design features, IRIS should provide an attractive option to utilities.

5.7.10 Project status and planned schedule

The IRIS project started in late 1999. It has completed the trade-off studies and conceptual design and is currently in the preliminary design stage. The project is already enough advanced that in October 2002 it has started its licensing process. Currently activities with the U.S. NRC are limited to a focussed pre-application. Essentially in this first phase of the pre-application licensing, NRC will review the project to provide feedback on two items that are considered as long lead items and therefore critical to the overall project schedule. These two items are identification of necessary testing and assessment of the risk informed regulation approach.

Licensing activities will intensify following completion of the AP1000 design certification, expected in 2005. Therefore, the current plan is to submit an IRIS design certification application in 2005, with the objective of obtaining design certification in 2008/2009. Following certification, with a parallel first-time-engineering effort, and an expected construction period of three years for the first IRIS module, IRIS deployment could occur as early as 2012, and more realistically around 2015.

5.8 QS-600E/W (CHINA NATIONAL NUCLEAR CORPORATION, CHINA)

5.8.1 Introduction

The QS-600e/w is based on the design of the Qinshan phase II (2×600MWe PWR NPP with two PWR units), used for electricity and potable water co-generation. It is being developed by the Beijing Institute of Nuclear Engineering (BINE), China National Nuclear Corporation (CNNC).

The design criteria are as follow:

- China national nuclear safety and environmental protection laws, regulations, guidance, norms, standards;
- Industry standards of China;
- Widely accepted international norms and standards.

5.8.2 Description of the nuclear systems

5.8.2.1 Primary circuit and its main characteristics

The reactor coolant system (RCS) consists of 2 loops with the capacity of 300MW per loop. Figure 5.8-1 shows the schematic diagram of its primary circuit.

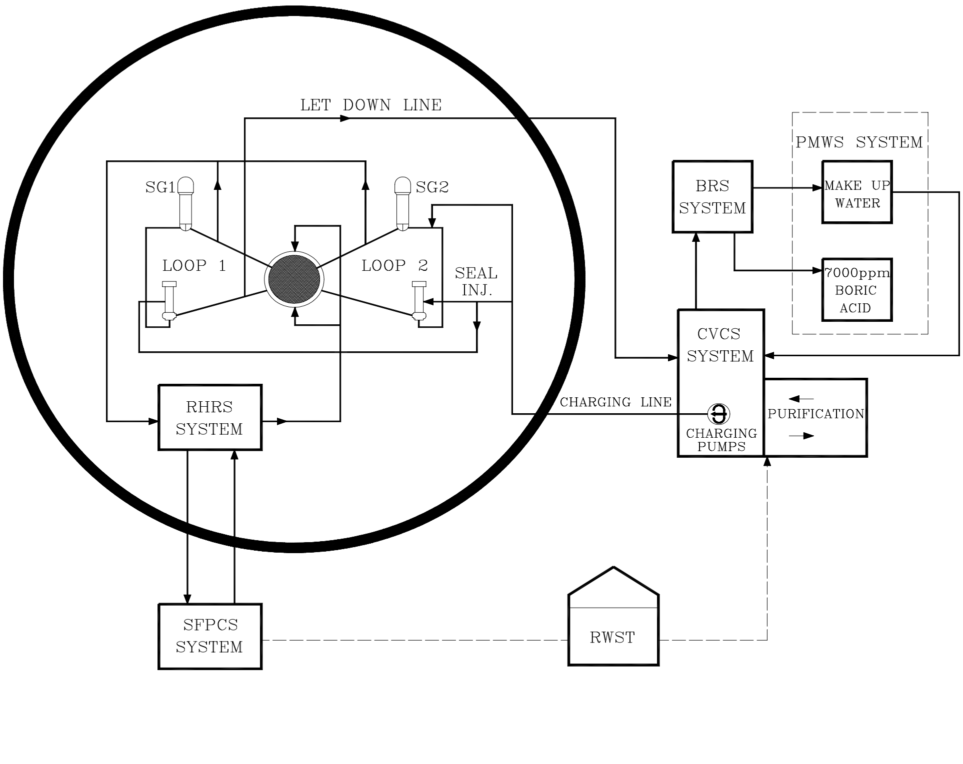


FIG. 5.8-1. Primary circuit of the coolant system

5.8.2.2 Reactor core and fuel design

The reactor core consists of 121 fuel assemblies. The designed fuel cycle is 12 months.

Core Height:	3658mm
Core Diameter:	2670mm
Fuel assembly type	17×17 AFA
Fuel material	Slightly enriched UO_2
Clad material	Zircaloy-4
Ave. Liner power density	15.99 kW/m
Initial enrichment	3.1/2.6/1.9
Ave. Burn-up	35000 MWd/tu

5.8.2.3 Fuel handling and transfer systems

During reactor shutdown and refueling, some of the reactor pool water is delivered into the storing water pool, and remain depth of the reactor pool, and then the spent fuel assemblies are replaced by fresh fuel.

5.8.2.4 Primary components

Primary components consist of the reactor vessel, the steam generators, the reactor coolant pumps, the pressurizer, the pressurizer relief tank, the reactor coolant pipes, valves, and so on.

5.8.2.5 Reactor auxiliary systems

Reactor auxiliary systems include chemical and volume control system, residual heat removal system, reactor boron and water make-up system, reactor cavity and spent fuel pit cooling and treatment system, steam generator blowdown system, etc.

5.8.2.6 Operating characteristics

Reactor Core Rated Thermal Power (MWt)	1936
Gross Electrical Power Output (MWe)	644
Plant Gross Efficiency (%)	33.3
Reactor Coolant Flowrate (m^3/h)	46640
RCS Operation Pressure (MPa)	15.5
Reactor Inlet/Outlet Temperature (℃)	292.8/327.2
Main Steam Pressure (MPa)	6.71
Linear Power Density (W/cm) Avg	159.9
Average Burn-up (MWd/tu)	35000

5.8.3 Description of the turbine generator plant and the desalination plant

5.8.3.1 Turbine generator plant systems

No information provided.

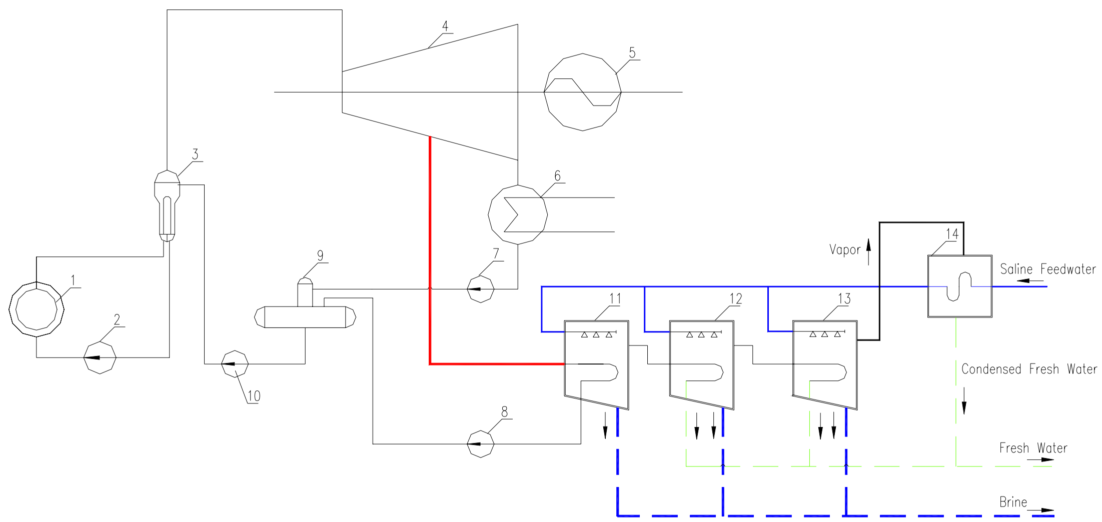
5.8.3.2 Description of desalination plant

5.8.3.2.1 Description of desalination process

The Multi-effect distillation (MED) process is used in this electricity and water co-generation plant. The MED facilities receive the steam from turbine extraction and produce fresh water. Figure 5.8-2 shows a schematic flow diagram of MED process.

The MED process is a kind of mature technology with a long application history. It is popularly used for the concentration of chemicals and food productions, and also is the first applicable process used for producing large scale of desalted water from sea water.

The feed sea water is heated by the steam in the first effect, resulting in evaporation of a fraction of the water content. The feed may be inside the heat transfer tube and the steam outside or vice versa. The concentrated brine is sent to the second effect maintained at slightly lower pressure than the previous effect. The vapor produced in the first effect condenses on the heat transfer tubes in the second effect, giving up its latent heat and generating an almost equal amount of vapor from the brine. The process is repeated from effect to effect at successively lower pressures. The condensate is collected as product water. The MED process produces around 5-25 ppm TDS (Total Dissolved Solid) product water quality from 35000 to 45000 ppm of sea water. The energy efficiency of the MED plant increases by increasing the number of effects. The Gained Output Ratio (GOR) for MED is theoretically equal to the number of effects, but practically somewhat less, because of heat losses.



1. Reactor; 2. Reactor Coolant Pump; 3. Steam Generator; 4. Steam Turbine;
5. Electric Generator; 6. Condenser; 7. Main Condensate Pump; 8. Condensate Pump; 9. Deaerator an Feedwater Tank; 10. Feedwater Pump; 11. First Effect
12. Second Effect; 13. Third Effect

FIG. 5.8-2. Schematic flow diagram of HT-MED process

In this proposal, a high temperature or a low temperature version of the MED is adopted, which is gaining ground for medium capacity desalination plants, owing to following advantages:

- Lower energy consumption;
- Higher heat transfer coefficient;
- Compactness;
- High product water quality;
- Reduced pre-treatment;
- Higher GOR, etc.

In the unlikely event of a leak in the tube wall, the vapor leaks into the brine chamber, thereby avoiding the contamination of product water. Aluminum, copper based alloys and titanium are used for the tubes and tube sheets. Epoxy lined carbon steel and stainless steel are used for the shell.

5.8.3.2.2 Coupling of desalination system with power generation plant

5.8.3.2.2.1 General description

In the case of nuclear plants that co-generate heat and electric power, the steam can be bled off at suitable points in the secondary circuit of the power plant for use by the desalination plant. However, protective barriers must be included in all co-generation modes to prevent potential carry-over of radioactivity.

There are two types of co-generation mode:

- Parallel co-generation;
- Series co-generation.

In parallel co-generation, electricity is produced as co-product along with desalted water by diverting a part of the steam to the turbine to produce electricity in the conventional manner and part of the steam to the desalination plant. This configuration allows increased flexibility in energy use. However, the total energy consumption would be the same as if the steam for desalination and electricity had been produced separately.

In series co-generation, electricity is produced by expanding the steam first through a turbine with an elevated backpressure or part of the steam to a appropriate extraction point of the turbine, and then to the desalination process. This form of co-generation results in reduced total energy consumption as compared to parallel co-generation. From the thermodynamic point of view, it is useful to convert most of the steam enthalpy to mechanical/electrical energy in the turbogenerator before using it as a heating medium in a thermal desalination plant for producing desalted water. Raising the turbine backpressure increases the temperature of the heat available to the desalination plant but reduces the amount of electricity generated. Selecting appropriate extraction pressure improves the efficiency of the desalination plant. Therefore, in series co-generation the turbine backpressure or extraction pressure must be optimized relative to overall plant economics.

5.8.3.2.2.2 Safety requirements for the couplings

The safety of a nuclear desalination plant depends mainly on the safety of the nuclear reactor and the interface between the nuclear plant and the desalination system. It must be ensured that any load variation of steam consumption in the desalination plant would not cause a hazardous situation in the nuclear plant. There should be suitable provision for monitoring the radioactivity level in the isolation loop and desalination system.

The basic requirement to prevent radioactive contamination of the desalination plant and/or the atmosphere is of utmost importance in thermal coupling. At least two mechanical barriers and pressure reversal between the reactor primary coolant and brine must be incorporated.

5.8.3.2.3 Preliminary analysis of electricity and desalination capacity balance

The amount of fresh water produced from this kind of electricity and water co-generation plant is varied in accordance with the local conditions and requirements. The electricity production can be increased while the water production be decreased, and vise versa. The electricity and seawater desalination capacity balance is briefly shown in Table 5.8-I.

5.8.4 Instrumentation and control systems

5.8.4.1 Design concept, including control room

Important safety functions include: reactivity control, heat removal from the primary system, steam generator water balance, primary circuit water balance, radioactivity control, containment integrity, etc.

The man-machine interface and man-factor engineering are considered on the design of the main control room.

TABLE 5.8-I. ELECTRICITY AND SEAWATER DESALINATION CAPACITY BALANCE

Plant	Desalination capacity m3/day	Net saleable power MWe
QS-600e/w	0	610
	60,000	598
	120,000	585
	240,000	562
	480,000	509

5.8.4.2 Reactor protection and other safety systems

Design principles on reactor protection systems are as follows: to evaluate the protection parameters including DNBR (Departure from Nuclear Boiling Ratio) and LPD (Linear Power Density), to use two-out-of-four logic, to adopt diverse reactor trip mechanisms, to follow all standards for 1E class equipment, to make mostly confirming single failure principle, to consider diversity, etc.

5.8.5 Electrical systems

5.8.5.1 Operational power supply systems

The operational power system consists of two independent trains, each of which has 100% capacity.

5.8.5.2 Safety-related systems

Safety-related systems are served by multiple power supply systems including the operational power supply system with two independent trains, direct current power supply system, diesel generator system, etc.

5.8.6 Safety concept

5.8.6.1 Safety requirements and design philosophy

During normal condition and operational occurrence, radioactive material shall be confined by maintaining the integrity of all physical barriers in the defense in depth: fuel rod claddings, primary pressure boundary, and containment.

Radioactive material released from the primary circuit in design basis accident shall be confined by maintaining reactor building integrity.

The high-temperature Multi-effect distillation (MED) process is used in this proposal to correspond the nuclear steam supply plant.

5.8.6.2 Safety systems and features (active, passive and inherent)

Safety systems includes: emergency core cooling system, containment spray system, emergency feedwater system, etc.

The emergency core cooling system includes high head injection subsystem, mid head injection subsystem, and low head injection subsystem.

The containment spray system contains two independent redundant trains. Each train includes 1x100% spray pump, 1x100% heat exchanger, connection pipes and nozzles.

The emergency feedwater system contains two independent redundant trains. Each train includes 1 motor driven pump, 1 turbine driven pump. Each pump has 100% capacity.

5.8.6.3 Severe accidents (beyond design basis accidents)

No information provided.

5.8.7 Plant layout

Plant layout principles are as follows:

- Physical isolation,
- Pipe whipping protection,
- Internal and external missiles protection,
- Radioactivity protection,
- Flooding and fire protection,
- Easy to install and maintenance,
- Earthquake consideration, etc.

5.8.8 Technical data

• Reactor heat output (Core)	1936 MW
• Max. electrical power output	644 MW
• Design pressure	17.2 MPa
• Nominal operating pressure	15.5 MPa
• Design temperature	343 ºC
• Average operating temperature	310 ºC

5.8.9 Measures to enhance economy and maintainability

In order to improve the plant economics, the following measures are to be taken, and the plant economics evaluation targets are shown in Table 5.8-2.

- To simplify the design
- To adopt simplified, inherent, passive or other innovative design features. Valves reduced 15% to 20%.
- To improve the operation flexibility
- Load follow operation compensates for day/night cycle load variations. Within a 24-hour period, power is ramped down for a few hours to any power level above 30%FP and then brought back to reference power level.
- To reduce the cost of equipment and structures
- In order to reduce the cost, some equipment and structures are localization.
- To reduce the construction period
- Containment dome integrity sling and installation, the construction period reduced to 48 months.
- To reduce the scope of maintenance during operation and outages
- To make the maintenance easier and with lower radiation exposure
- Layout principles are: Physical Isolation, Pipe Whipping, Internal and External Missiles, Radioactivity Protection, Flooding and Fire Protection, Easy to Install and Maintenance, Earthquake Consideration
- To increase the plant availability and load factor
- The plant availability and load factor is ⩾87%
- To reduce the power generation cost

All measures are to be taken above only general consideration during this concept design. These will be described more in detail design.

5.8.10 Project status and planned schedule

No information provided.

5.9 PAES-600 FLOATING TWIN UNIT NPP WITH VBER-300 REACTOR (OKBM, RUSSIAN FEDERATION)

5.9.1 Introduction

To improve the performance of the pilot floating NPP with the KLT-40S reactor for Severodvinsk, modular plants with increased power are being currently developed, including an 850 MW(th) capacity modular reactor design. The work performed has shown a possibility to create 590 MW(e) output twin-unit floating NPP on the basis of the modular reactor.

The objectives of the floating NPP with VBER-300 reactor are the following:

1. Creation of a mobile, economically efficient, high capacity power source for coastal regions.

2. Provision of guaranteed safety through the defence-in-depth approach, self-protection ability of the reactor and the use of passive safety systems.

3. Improvement of economics through increase in the capacity of a power unit.

4. Implementation of proven nuclear propulsion technologies for electricity and heat supply purposes.

5.9.2 Description of the nuclear systems

5.9.2.1 Primary circuit and its main characteristics

The following main principles are laid in the design basis of the VBER-300 reactor plant for a floating NPP:

- An 850 MW(th) VBER-300 reactor plant is implemented, which is developed on the basis of proven modular-type propulsion PWR technologies that have been successfully operated in Russia;

- Long-term experience in design, validation, fabrication and operation of similar nuclear reactors is taken into account (about 6000 reactor-yr. total operating record);

- Compact NSSS configuration enables minimization of capital costs for the floating power unit construction;

- Highly reliable systems and equipment used in propulsion power plants are implemented;

- Implementation of WWER-type PWR operating experience and proven technical features, as well as AST-500 nuclear district heating reactor development experience;

- Enhanced safety level is ensured, the safety requirements for new generation NPPs (IAEA recommendations) are met;

- Minimal radiological impact to personnel, public and the environment, so that consequences of any accident are limited to the floating power unit area;

- Economic efficiency and competitiveness;

- Previously accomplished R&D results are used in maximum feasible extent to support the design; no need for large scale research work; developmental work is only needed for individual equipment;

- Standardized design (i.e. the reactor plant may also be used for a land based NPP), that enables minimization of the design work costs and the design quality level to be enhanced.

Fig. 5.9-1 shows the VBER-300 reactor module design basis.

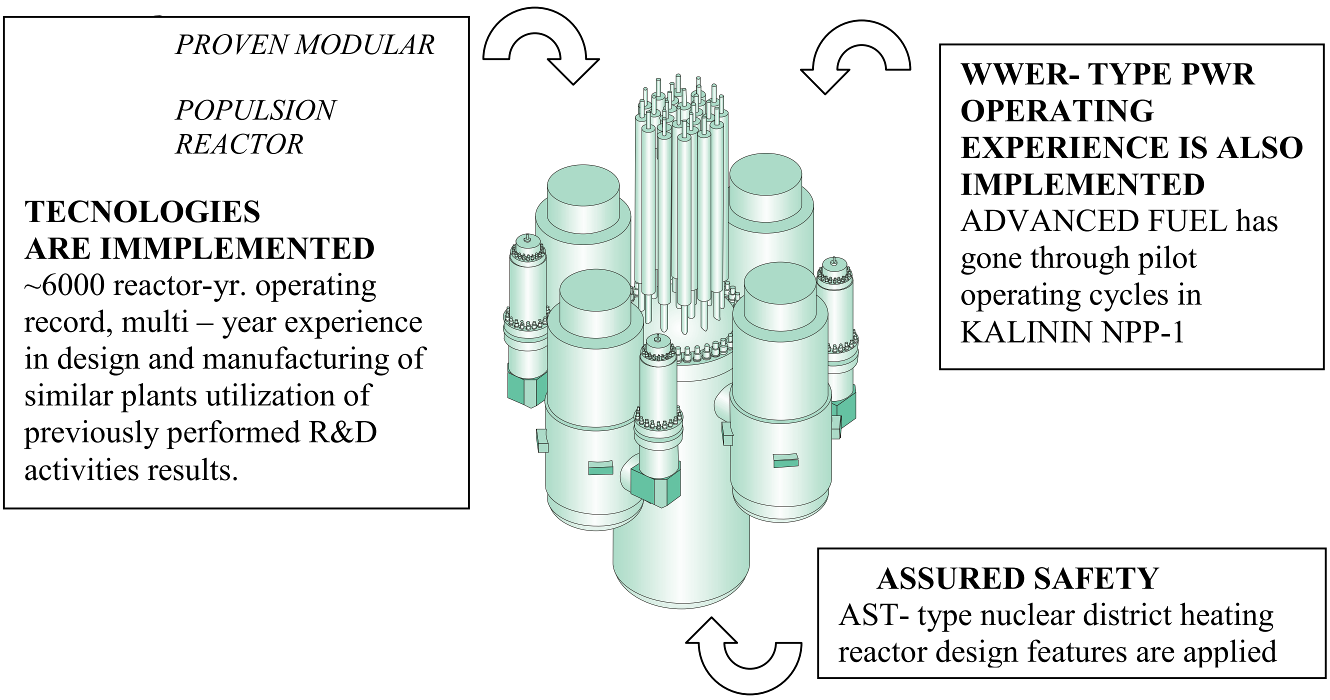


FIG. 5.9-1. VBER-300 reactor module design basis.

The primary (pressurized water) coolant is circulated in the module through four heat transport loops.

The VBER-300 reactor module integrates:

- Vessel unit (the reactor vessel, steam generators and RCPs are connected by short fitting pipes),
- Active core,
- Once-through steam generators (SGs),
- Main leak-tight circulating pumps (RCPs),
- Control and protection system (CPS), drive mechanisms.

The modularized design of the reactor plant enables minimization of a nuclear island building volume and consequently specific capital cost, as well as elimination of primary piping and hence both large and medium LOCAs. The maximum potential size of a LOCA is equivalent to 32 mm.

The key items of the reactor module equipment are designed on the basis of those serially produced and operated in propulsion nuclear steam supply systems. The fault free operating record of this type of equipment currently exceeds 150 000 h.

The VBER-300 reactor module uses once-through coiled cassette-type steam generators with secondary coolant at the tube side. This type of SG has a very compact configuration, high reliability and repairability. A centrifugal glandless pump with a canned motor is used as RCP.

Table 5.9-I summarizes the main data of the primary circuit:

TABLE 5.9-I. VBER-300 REACTOR PRIMARY SYSTEM CHARACTERISTICS

Parameter	Value
Thermal capacity, MW	850
Pressure, MPa	15.7
Core inlet/outlet temperature	294/332
Coolant flow, t/h	13610

5.9.2.2 Reactor core and fuel design

The VBER-300 reactor module implements a cassette-type active core with OKBM-developed advanced fuel sub-assembly (AFSA) that uses a rigid skeleton and has a number of positive features (e.g., improved operational and thermo-technical reliability). The fuel sub-assemblies have been successfully tested by four-year operation at Kalinin NPP unit No1 with WWER-1000 reactor. Basic core design data are given in the Table 5.9-II.

TABLE 5.9-II. ACTIVE CORE MAIN CHARACTERISTICS

Parameter	Value
Number of AFSA	85
Power density, MW/m^3	77
Max linear heat rate, W/cm	242
Reloading regime	3x1.5 yr.
Uranium load in equilibrium mode, t	10
Max. fuel burnup, MWd/kgU	50
Number of control rods	61

The active core key design features are the following:

- Pelletized uranium dioxide fuel is used, that is completely licensed and tested in WWER-type reactors, ensures high uranium burnup, moderate power transients;
- Only zirconium alloys are used within active portion of AFSA, thus neutron losses are minimized;
- Gadolinium burnable poison integrated in the fuel is used in a form of rods;
- Highly efficient reactivity compensation mechanical system ensures the reactor core subcriticality even at a stuck rod event with the most reactive control rod, while there is no boric acid in coolant;
- Reduced power density, optimized water-to-uranium ratio.

The VBER-300 reactor plant uses WWER-1000-type fuel rods with uranium dioxide of 5% enrichment.

5.9.2.3 Fuel handling and transfer systems

New fuel is delivered to the plant storage in casks transported by a special depot-ship. The new fuel storage capacity corresponds to a total amount of fuel subassemblies in the reactor core with 20% margin. The new and spent fuel storage is located in the central pontoon between the reactor plants.

Spent nuclear fuel is stored in a special pool for at least 3 years and then (once in a five-year period) is removed from the plant by a depot-ship. The shipping cask with spent fuel is handled by a bridge crane of 125/20 t capacity.

Solid and liquid radwastes produced at the plant are also removed by a depot-ship simultaneously with spent fuel.

5.9.2.4 Primary components

No information provided

5.9.2.5 Reactor auxiliary systems

No information provided.

5.9.2.6 Operating characteristics

No information provided.

5.9.3 Description of the turbine generator plant systems

5.9.3.1 Turbine generator plant

The floating NPP design with VBER-300 reactors envisages application of AOOT “LMZ”-designed T-275/200-60/50-type turbine plant with a heat extraction plant of about 460 Gcal/h capacity. The turbine plant consists of one high pressure and one low pressure cylinders. The turbine design is based on proven features that are used at operating NPPs with WWER-type reactors, but they are naturally adapted for operating conditions specific for floating installations.

5.9.3.2 Condensate and feedwater systems

No information provided

5.9.3.3 Auxiliary systems

No information provided

5.9.4 Instrumentation and control systems

Information on the plant’s I&C systems will be presented at the systems design stage.

5.9.5 Electrical systems

Information on the plant’s electrical systems will be presented at the systems design stage.

5.9.6 Safety concept

5.9.6.1 Safety requirements and design philosophy

The VBER-300 reactor module safety concept is based on the following principles:

- Defence-in-depth safety,
- Inherent self-protection ability,
- Passive safety systems and devices are implemented,
- Tolerance to abnormal external impacts corroborated by practice.

Accident prevention and mitigation are ensured by:

- Negative values of reactivity coefficients on the reactor power, fuel temperature, coolant density and temperature within the entire range of parameters variation;
- Modular configuration of the reactor plant, flow restrictors eliminate large and medium size LOCAs and simplify emergency core cooling system (max break size in primary system is as small as 32 mm);
- Self-actuated devices to initiate safety systems;
- Reactor pressure vessel emergency cooling system, thus eliminating the need for core debris trap.

Table 5.9-III summarizes basic safety characteristics of the reactor.

The PAES-600 NPP has the features essential for its environmental and radiological safety:

- Leak-tight primary system that eliminates the radiological impact to the public and environment;
- Insignificant consequences of any accidents limited to the plant floating platform.

TABLE 5.9-III. SAFETY CHARACTERISTICS

Parameter	Value
Negative values of reactivity coefficients on fuel temperature, coolant temperature, reactor power	
The mechanical reactivity compensation system ensures the reactor sub-criticality under cooled de-poisoned condition without boric acid in primary coolant and with the most reactive control rod stuck	
Reduced core power density	76.8 kW/l
Max break size in primary system	≤ 32 mm
Duration of passive systems functioning in LOCA accidents	72 h
Duration of passive systems functioning in black-out accident	72 h
Severe core damage probability	$<10^{-8}$ per r.-yr.

5.9.6.2 Safety systems and features (active, passive and inherent)

Reactor emergency protection (shutdown) system

- Mechanical shutdown system -absorber rods are moved down under gravity as drive mechanisms are de-energized to scram signals or special passive devices (pressure operated circuit breaker) actuate if automatic systems fail;

- Liquid boron injection system - boric acid solution is injected into the reactor by make-up pumps.

Residual heat removal

- Passive emergency shutdown cooling system - by natural circulation of coolants in all heat transport circuits with stored water evaporation; self-actuated devices initiate if automatic equipment fails;
- Emergency heat removal system via the coolant purification system's heat exchanger;
- Heat removal by secondary circuit with the use of steam turbine plant equipment.

Emergency core cooling

- Passive water injection system with accumulators operated by compressed gas;
- Active water injection system with recirculation - ensures long-term cooling the reactor core in LOCA;
- Active primary circuit make-up system.

Confinement barriers system

- Fuel cladding;
- Hermetically-sealed primary circuit;
- Fast-response valves in piping of primary and secondary circuits and in equipment cooling circuit;
- Protective shell (leak-tight steel containment and protective enclosure with filtration of releases);
- Reactor pressure vessel cooling system.

5.9.6.3 Severe accidents (beyond design basis accidents)

No information provided.

5.9.7 Plant layout

5.9.7.1 Buildings and structures, including plot plan

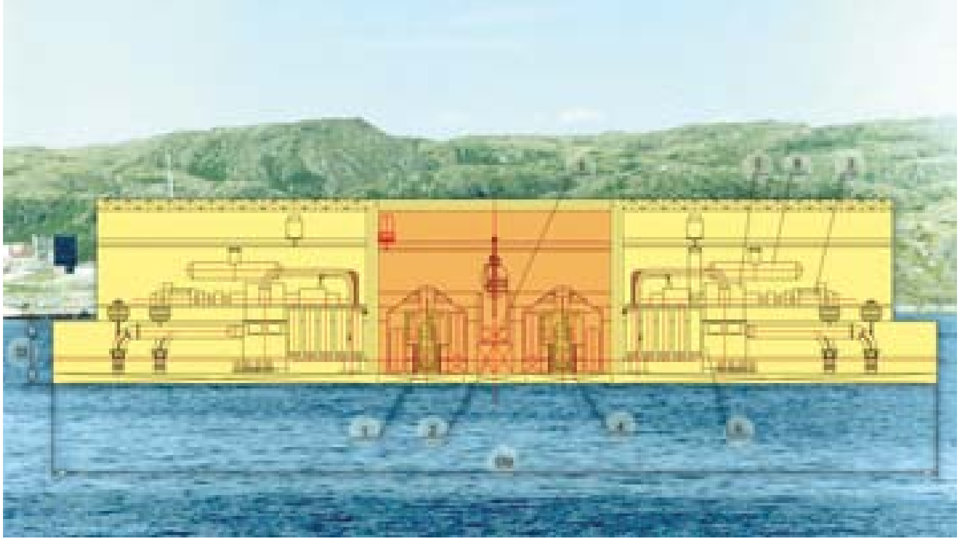
The PAES-600 plant layout takes into account the need for its physical protection against both internal and external impacts.

The plant includes the following structures: a floating power unit, a depot-ship, protective dam, mooring wall, 220 kV power transmission line, plant area fencing, 12 m-15 m depth protected water area, non-residential land area (about 30 000 m^2), car road with loading/unloading sites.

5.9.7.2 Reactor building

The NPP floating power unit (Fig. 5.9-2) is a non-self-propelled autonomous floating structure related to the pillar-class ships according to Russia's Sea Navigation Register classification. The floating power unit is located on a platform consisting of three pontoons (one central and two peripheral ones).

The independent reactor plants are in-line located in the central pontoon. Each reactor plant consists of the following divisions: reactor division, power plant control board and electrical equipment divisions; areas for the plant refuelling and repair are also provided.



1- Reactor plant-1; 2- Radwaste storage facility; 3- New and spent FSA facility; 4- Reactor plant-2; 5- Generator; 6- Condenser; 7- Steam turbine; 8- Deaerator

FIG. 5.9-2. Floating NPP layout.

A steel protective shell (containment system) houses the main equipment of the reactor module with related service systems.

A fuel storage is located in the central pontoon between the reactor plants.

Turbine generators and their related equipment and systems are located in the stern and fore sections of the central pontoon.

Electrical equipment for power transformation, distribution and supply (at up to 220 kV voltage) to coastal objects and that for power supply to the plant house loads are located in a left board pontoon.

The right board pontoon houses auxiliary equipment, i.e. back-up and emergency power sources, pumps, etc.

Each pontoon has the following dimensions:

- length, m – 170
- width, m – 19
- side height, m – 12.

Total displacements of the floating power unit is 49 000 tons.

A composite steel-ferro-concrete vessel of the power unit eliminates the need for scheduled docking during the plant lifetime. The period between overhauls (20 years) depends on the key reactor plant equipment life, which is about 150 000 h. The PAES-600 NPP operating life is 60 years.

5.9.7.3 Containment

No information provided

5.9.7.4 Turbine building

No information provided

5.9.7.5 Other buildings

No information provided

5.9.8 Technical data

TABLE 5.9-IV. VBER-300 FLOATING POWER UNIT CHARACTERISTICS

Parameter	Value
Electric output, MW	2x295
Reactor thermal power, MW	850
Superheated steam:	
- pressure, MPa	6.38
- temperature, ºC	305
Capacity factor	0.9

5.9.9 Measures to enhance economy and maintainability

The plant economic effectiveness is ensured through implementing the following features:

- Flexible technology of construction, transportation to a NPP site and decommissioning; possibility of floating power unit disposal at a specialized plant;
- High reliability corroborated by positive operating experience with prototype reactors;
- Very compact configuration, low specific expenses of the construction material (metal), minimized capital cost per kilowatt;
- Application of proven propulsion reactor technologies.

The most important economic indicators of the floating NPP are given in Table 5.9-V.

Total duration of the twin-unit plant design and construction is estimated to be 7 to 8 years, the plant construction period is 4 years.

5.9.10 Project status and planned schedule

Conceptual design is under development.

TABLE 5.9-V. FLOATING NPP MAIN ECONOMIC DATA

Parameter	Value
ELECTRIC CAPACITY, MW	2x295
Service life, yr.	60
Construction cost, M$	470
Specific capital cost per kWe, $	800
Expected generating cost, cent/kW·h	1.8
Capital cost recovery period since FNPP commissioning, yr.	≤ 7

5.10 NP-300 TECHNICATOME, FRANCE

5.10.1 Introduction

Technicatome’s NP-series design relies on technologies derived from years of maritime propulsion experience and the same fuels common in large capacity electricity-generating plants.

This French-developed concept, using time-proven technologies — where possible in association with local suppliers — builds on the PWR expertise gained worldwide and notably in France, where, for more than 20 years, Electricité de France (EDF), CEA and their industrial partners have engineered their present commercial nuclear power network.

TECHNICATOME/AREVA has, for over 40 years, has been the prime contractor for the strategic nuclear propulsion systems used by the French Navy. This involved developing small-sized nuclear boiler plants of the same design as the Nuclear Steam Supply Systems (NSSSs) used in France's civilian NPPs. The 19 units already built for maritime propulsion represent about 300 reactor-years of experience without a single noteworthy operating incident. 12 such units are currently in service and the same number are slated for commissioning by 2010.

TECHNICATOME/AREVA's experience in designing, building and operating PWR power plants in France and other countries forms the basis for the design of the medium size NP series. In the case of export projects, TECHNICATOME/AREVA plans to act as prime contractor for plant construction, with subcontracted work to local suppliers with the required competencies.

The NP series has a basic component lifetime of more than 50 years.

The optimum size of an electrical generating plant depends essentially on the grid to which it is connected. In general, grid stability — a key element in reliable supply of electricity to consumers — is obtained by multiple, limited-size generating plants that are suitably disseminated over the network.

Based on these observations, TECHNICATOME/AREVA and its partners decided that the best compromise for successfully competing with coal or gas and responding to the forecasted development of many of the world's grids, is to provide a choice of plants within a nominal power range from 120 to 450 MW(e).

The NP-300 is designed for maximum operational flexibility (fast response to changes in grid demand or "load follow").

5.10.2 Description of the nuclear systems

The standard NP-300 essentially consists of a nuclear part with a pressurised light water reactor providing 1000 MW(th) in the form of saturated steam, and a conventional electricity-producing part equipped with a 300 MW(e) turbine-generator unit.

5.10.2.1 Primary circuit and its main characteristics

Developed by TECHNICATOME/AREVA in the early '70s for merchant vessel propulsion systems, the CAS (**C**haufferie **A**vancée de **S**érie) concept consists of a single-block boiler plant in which each steam generator is linked to the reactor by a short, straight connection carrying concentric hot and cold primary coolant flows and pumps are integral with the steam generator (SG) channel heads.

The vessel/SG connection is considered "safe" in the same way as the vessel closure system. Its failure is not credited in safety design calculations. Note, however, that use of internal flow separators in this line and a nozzle reinforcing its resistant structure, would limit leakage flow in the event of a break.

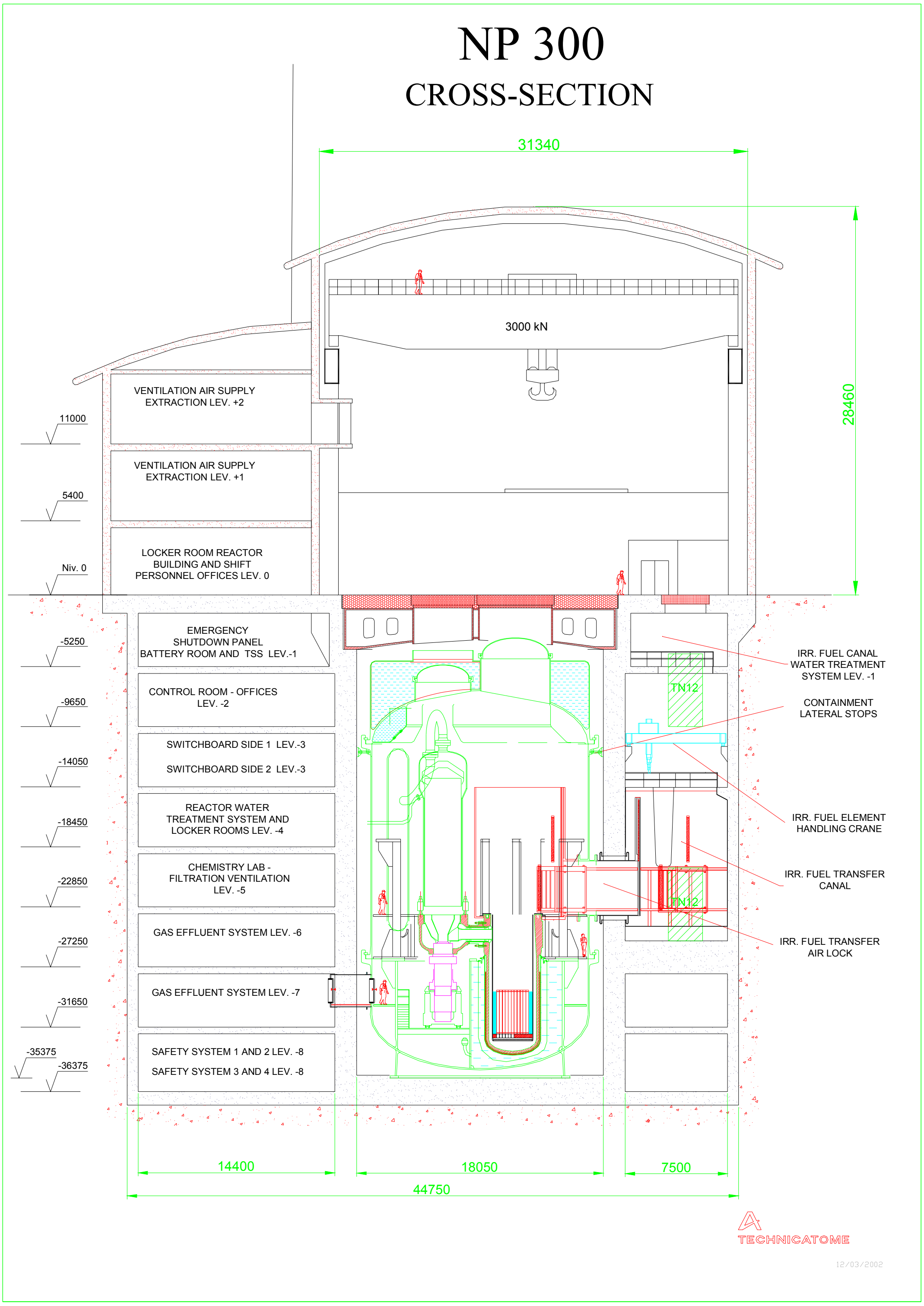


FIG. 5.10-1. NP-300 cross section.

The complete unit is installed on a single-piece support frame that allows equipment to expand freely within the strict limits required to preserve its geometry under all conditions, including earthquakes.

Both NP-300 steam generators are flanged to the channel head. SG tubes are the U-type mounted in a concentric arrangement.

This concept is applicable to plants generating up to 1500 MW(th) and was selected for the NP-300 line for the advantages offered by its compact primary system geometry, i.e.:

- Simplified onsite assembly of the primary system and easier assurance of system quality;
- Minimisation of containment volume, due to primary system compactness;
- Simplified system inspection and easier performance of SG replacements, if necessary.

A number of basic CAS system concepts were likewise applied to the CAP (Advanced propulsion reactor) that has been operating at Cadarache since late 1975. This facility has enabled TECHNICATOME/AREVA to partner with EDF, under the French NPP program, for a wide range of experiments, specifically on fuel.

5.10.2.2 Reactor core and fuel design

a) General description

- The fuel is uranium di-oxide enriched to the usual levels (<5%) for commercial nuclear power plants. Plutonium di-oxide can be used as an option;
- One-third of the core is replaced every 18 months to 2 years, with refueling scheduled during periods of low demand. 1/4 core refueling is a possible option;
- The fuel is used in rod form;
- The core is "open";
- The fuel elements are of the type used for electricity generation in France, except for the rod length;
- Absorbers rods are of the standard gray or black "cluster" type used for French electricity generation;
- Control by boric acid solution is possible ***under strict conditions and limitations***. An option without soluble boron under normal operation is to be examined;
- Burnable poisons in the form of clusters or mixed in the fuel are used (boron, hafnium, gadolinium, erbium, dysprosium). The core contains 109 elements (maximum) with an active height of 3000 mm (maximum) (the active height can be optimized within the range 2500 mm to 3000 mm);
- The core contains one central element — and therefore an uneven number of elements — so that mechanisms can be installed in a checkerboard pattern with 1/4 core symmetry;
- The core is equipped with a stainless steel heavy reflector;
- The core contains nuclear and thermal instrumentation;
- The fuel elements are of the "sealed" type, i.e. work cannot be carried out on the fuel elements (to replace them for example) without leaving an indication that this has been done. This assists the IAEA safeguards inspection.

b) "Enriched uranium" option

- The equilibrium enrichment is 4.5% for the 1/3 core refueling option, and is to be defined for the 1/4 core refueling option;
- The discharge enrichment (in U^{235}) is roughly 1.5%;
- The average fuel burn-up achieved is 33000 MWd/t of feed uranium for the 1/3 core refueling option and potentially 44000 MWd/t for the 1/4 core refueling option;
- The plutonium produced by irradiation of uranium fuel (initially greater than 95% U^{238}) has a Pu^{240} content of at least 20%.

c) "MOX" option

The MOX option guarantees that the minimum proportion of Pu^{240} is present in the fuel at any moment of the core life.

5.10.2.3 Fuel handling and transfer systems

a) Refueling system (Fig. 5.10-2)

- The fuel is removed from the side via an air lock and transfer canal built into the reactor block and protected against hazards;
- The fuel removal and renewal system is fully interfaced with the transfer systems used in French power plants;
- All interim storage and fresh or irradiated fuel transfer areas are monitored remotely by cameras that can be used for IAEA safeguards inspection. These systems allow all operations carried out on fresh or irradiated fuel to be monitored.

b) Handling mechanisms for reactivity control rods

- The mechanisms are derived directly from standard mechanisms used in French electricity production;
- They are arranged in a checkerboard pattern with 1/4 core symmetry containing the central element. However, in the preferred option the central element is not borated (this produces "natural" and improved balancing of cluster efficiency);
- The mechanisms used (MAGNETIC JACK type) are the same as those in French electricity generating reactors of the N4 plant series and EPR ; in particular the operating stroke is the same, so that operating and drop safety conditions are unchanged.

Given the low active height of the core, the additional available stroke allows the upper level of the fuel to be installed as low as possible in the vessel.

5.10.2.4 Primary components (Figures 5.10-3(a), 5.10-3(b), 5.10-3(c))

Reactor vessel (Fig. 5.10-3(a))

- The vessel has an elongated or "bottle" shape, with a large volume of water located above the core (note that the "raised" position of the core in the vessel depends on the types of mechanism chosen).
- The reactor vessel closure head is a flat cover.
- Fluid penetrations are systematically positioned as high up as possible via the vessel flange (and possibly via the closure for a small number of penetrations).
- There is no penetration in the vessel bottom head, and no pipe capable of siphoning the vessel in the event of an accident.
  - These straightforward measures allow the level of primary breaks to be placed "as high as possible", a factor taken into account in the safety analysis.
- The vessel is protected from fast neutron radiation by:
  - a large water gap due to its "bottle" shape,
  - layers of steel in the form of:
    - the heavy reflector ("heavy baffle"),
    - the barrel shell providing additional protection.
- The vessel + steam generator assembly is supported on trunnions or bosses that are weld-assembled to flanges. All these elements are in the same plane.
- The vessel is fitted with an external passive cooling system capable of confining the corium formed in the event of a serious core meltdown accident inside the vessel.

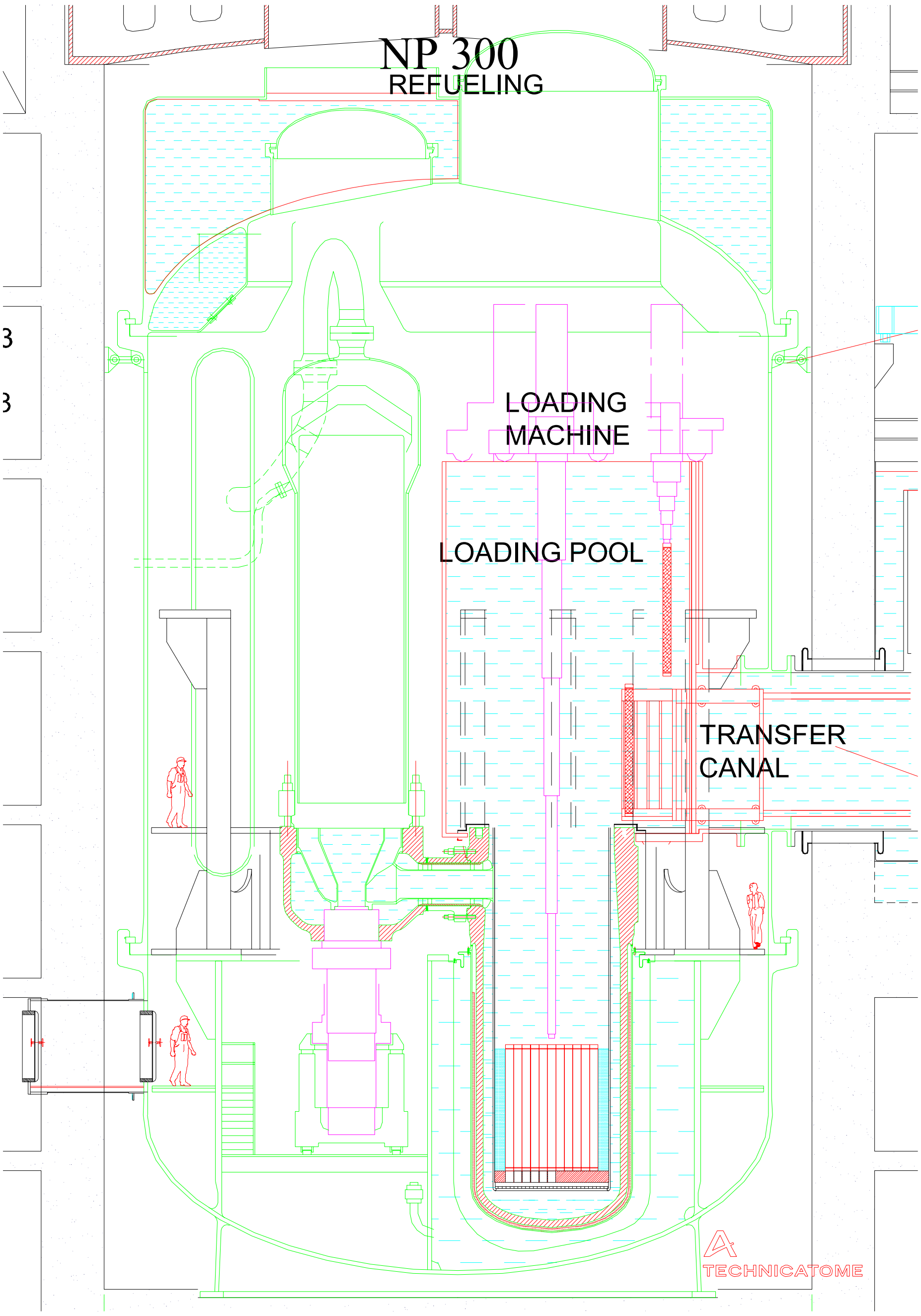


FIG. 5.10-2. NP-300 Refueling system layout.

NP 300

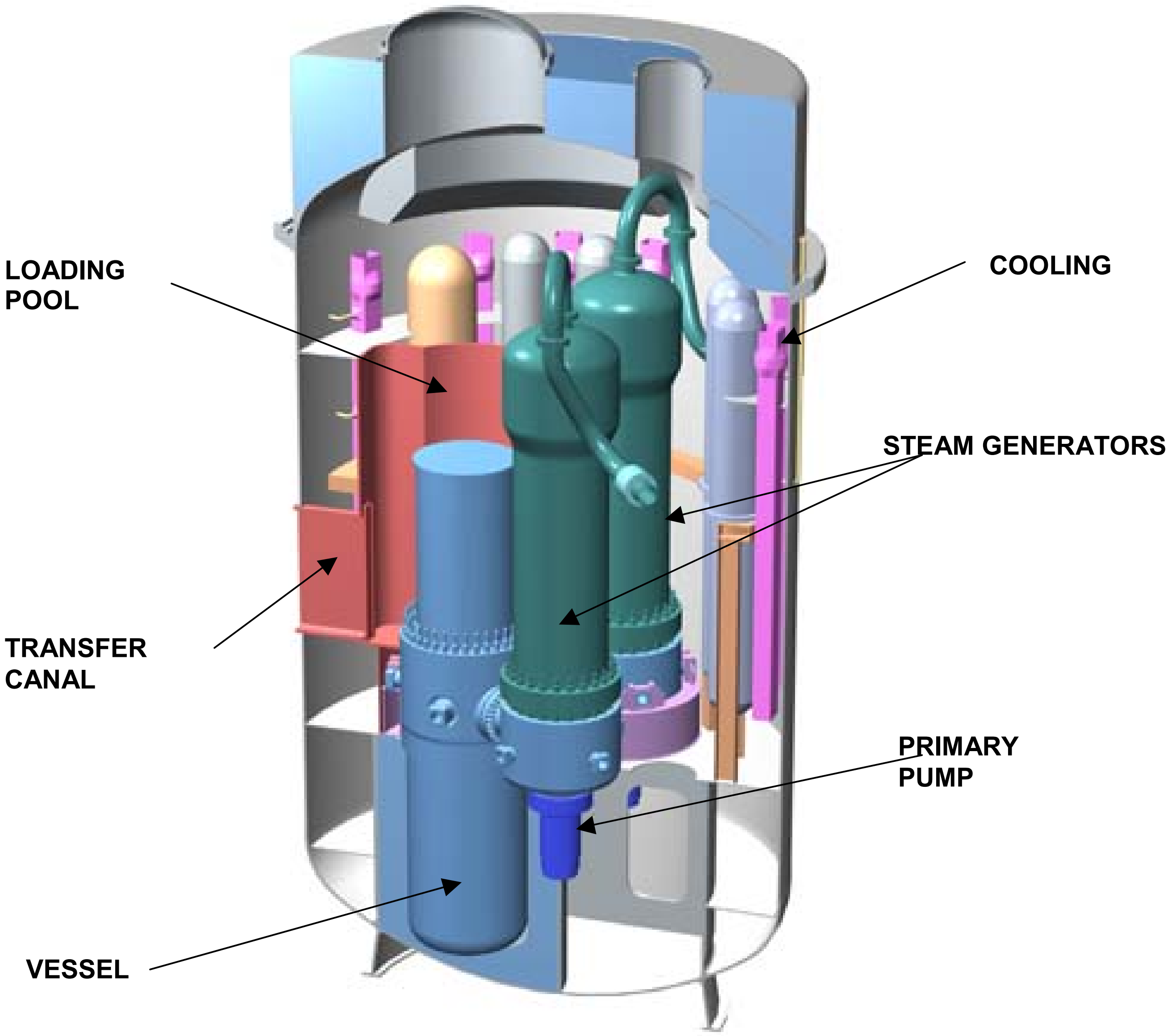


FIG. 5.10-3(a). General view of the primary circuit: reactor vessel.

Steam generator (Fig. 5.10-3(b))

- U-shaped tubes derived from those in use in French power plants.
- Steam produced: saturated, 34.7 bar
- The steam produced may be slightly wet. Steam drying is conducted outside the containment.

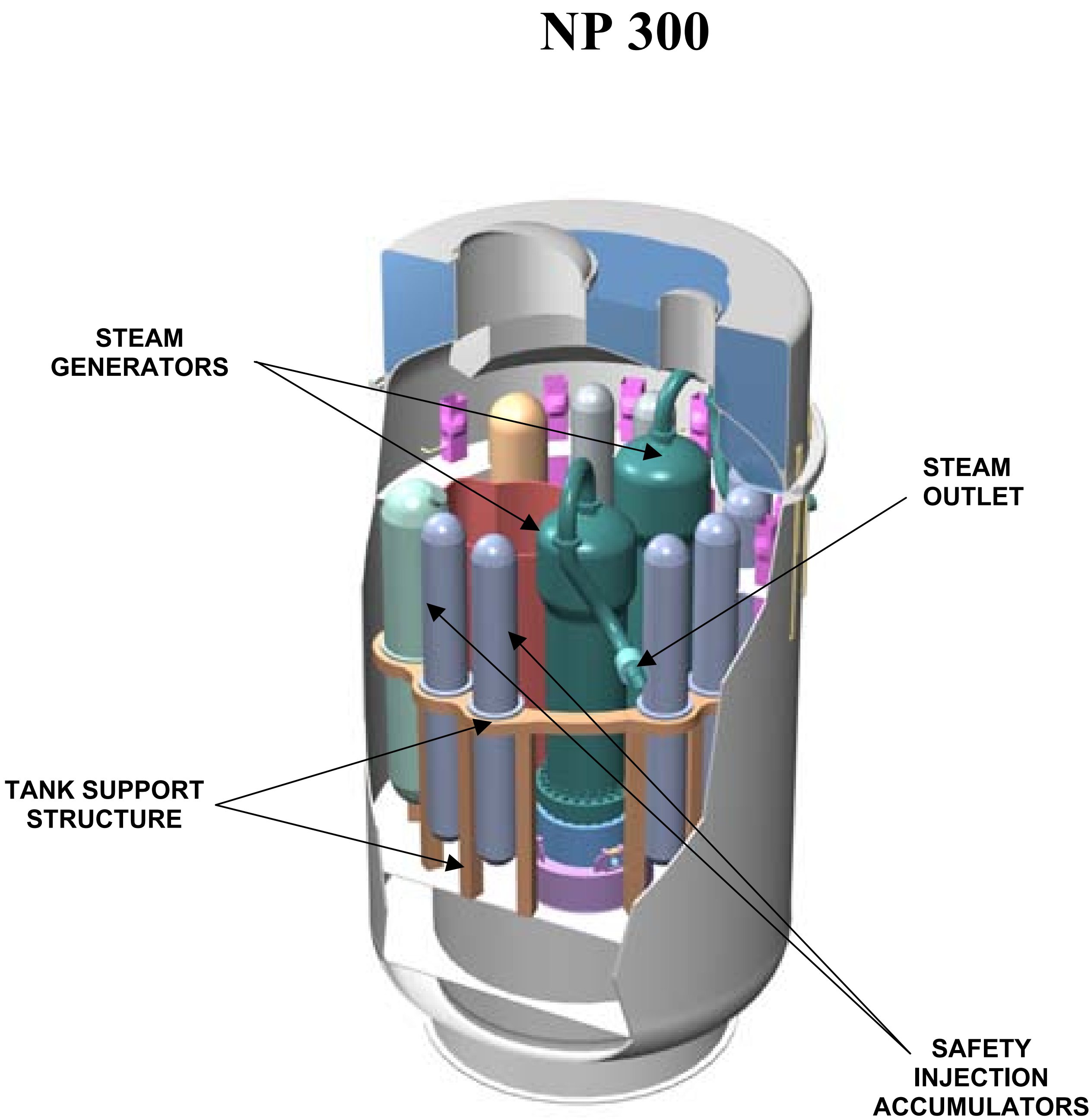


FIG. 5.10-3(b). General view of the primary circuit: steam generators.

Pressurizer – Reactor-pressurizer link – Primary pressure relief valves (Fig. 5.10-3(c))

The pressurizer is of the "standard" type with a self-regulating pressure device. It is connected to the primary circuit via a continuously rising pipe.

Main primary pumps

- 2 pumps each installed below a steam generator ("upside down");
- Asynchronous canned-type motor;
- The "upside down" arrangement assists pump operation in the following manner:
  - venting occurs naturally,
  - natural convection of the water improves heating conditions and increases the efficiency of the thermal barrier;
- The primary pumps are slowed down under inertia.

5.10.2.6 Operating characteristics

All NP-300 facilities are controlled, operated and monitored remotely from a single control room.

These functions implement state-of-the-art digital technology. The human engineering and man/machine interface techniques are based on TECHNICATOME/AREVA's experience with naval propulsion facilities in a highly confined and harsh environment.

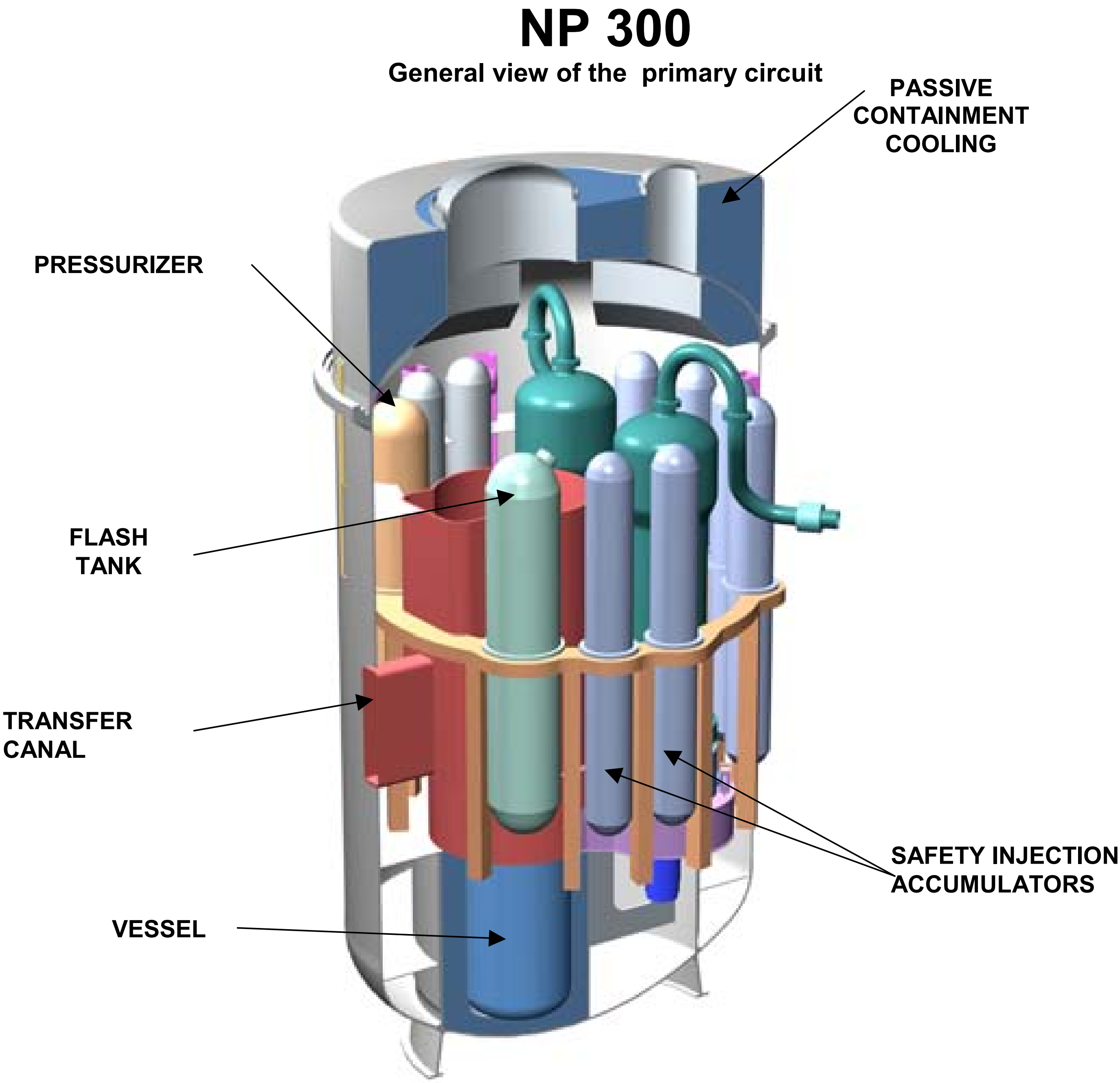


FIG. 5.10-3(c). General view of the primary circuit: reactor vessel – pressurizer link

5.10.3 Description of the turbine generator plant systems

5.10.3.1 Turbine generator plant

Turbine

The turbine comprises:

- A live steam inlet comprising one or more stop valves and control valves for the supply to the turbine cylinder;
- An MP cylinder;
- A low-pressure steam inlet comprising a steam isolation valve and a steam flow control valve;
- 2 double-flow low-pressure cylinders;
- 2 steam bleedoffs to supply the MP heaters;
- 3 steam bleeding points to supply the LP heaters;
- A steam overspeed and overpressure protection system;

The rotation speed of the turbine-generator unit is 1500 rpm.

Dryer / overheater condensor

The heating steam is taken directly from the steam manifold at the steam generator outlet, upstream of the turbine inlet valves. The wet steam from MP expansion is then dried and superheated before admission to the LP cylinder.

The dryer and superheater condensates are recovered in the deaerating feedwater tank.

Electrical generator

The turbine drives directly the rotor of a three-phase generator rotating at 1500 rpm and supplies electrical power to the step-up transformers. The electrical power is transmitted to the grid at a voltage of 400 kV (or 63 kV).

The generator is dimensioned for guaranteed nominal power of 320 MW with a nominal cosine ϕ of 0.9.

5.10.4 Instrumentation and control systems

5.10.4.1 Design concept

The control room and detailed installation proceduresare based on experience of TECHNICATOME/AREVA.

The management, control and radiological protection team of an NP-300 plant should comprise about 50 engineers and technicians.

5.10.5 Electrical systems

5.10.5.1 Operational power supply systems

No information provided.

5.10.5.2 Safety-related systems

No information provided.

5.10.6 Safety concept (Fig. 5.10-4)

5.10.6.1 Safety requirements and design philosophy

A major nuclear program requires a regulatory basis for design and construction. France has adopted an essentially practical approach, initially based on the full set of American codes. Gradually, as intensive studies were conducted in France by French Safety Authorities, Electricité de France and constructors, these rules were refined and extended so that there now exists an exhaustive and consistent set of rules applicable specifically to PWRs.

Thus, with regard to the design of nuclear systems, the French rules are derived from the American codes with additional elements based on experience acquired from the nuclear power units in the French program. A time-proven body of design and construction specifications for electrical and mechanical equipment and civil structures has existed in France for some thirty years, and these were maintained for the nuclear program by the French Safety Authorities.

Proposed safety systems must always conform to IAEA safety standards, French standards and standards of the operator's country.

The design of the NP-300 plant requires total separation of the containment from the slab protecting it against external hazards.

The corresponding structures are designed to be easily adaptable to local conditions.

The nuclear part of the NP-300 is totally underground. The overlying protection slab is designed to withstand aircraft crashes.

Containment of fission products ultimately depends on intrinsic design characteristics. It is provided in the NP-300 plant by an oval-shaped metal containment.

However, the definition of external hazards (aircraft crash, etc.) may vary considerable from site to site. The corresponding protection is therefore provided in the NP-300 plant by burying the reactor containment, together with the mechanical and electrical systems essential for safety and the fuel handling and storage facilities. The cube-shaped "nuclear module" is specially adapted to construction in an excavated site. The strength of the protection slab can be adapted to the characteristics of external hazards identified for the site.

Handling of heavy, bulky parts inside the containment is also separated from both the above functions and is performed by an overhead crane traveling in a light structure above the module/slab assembly.

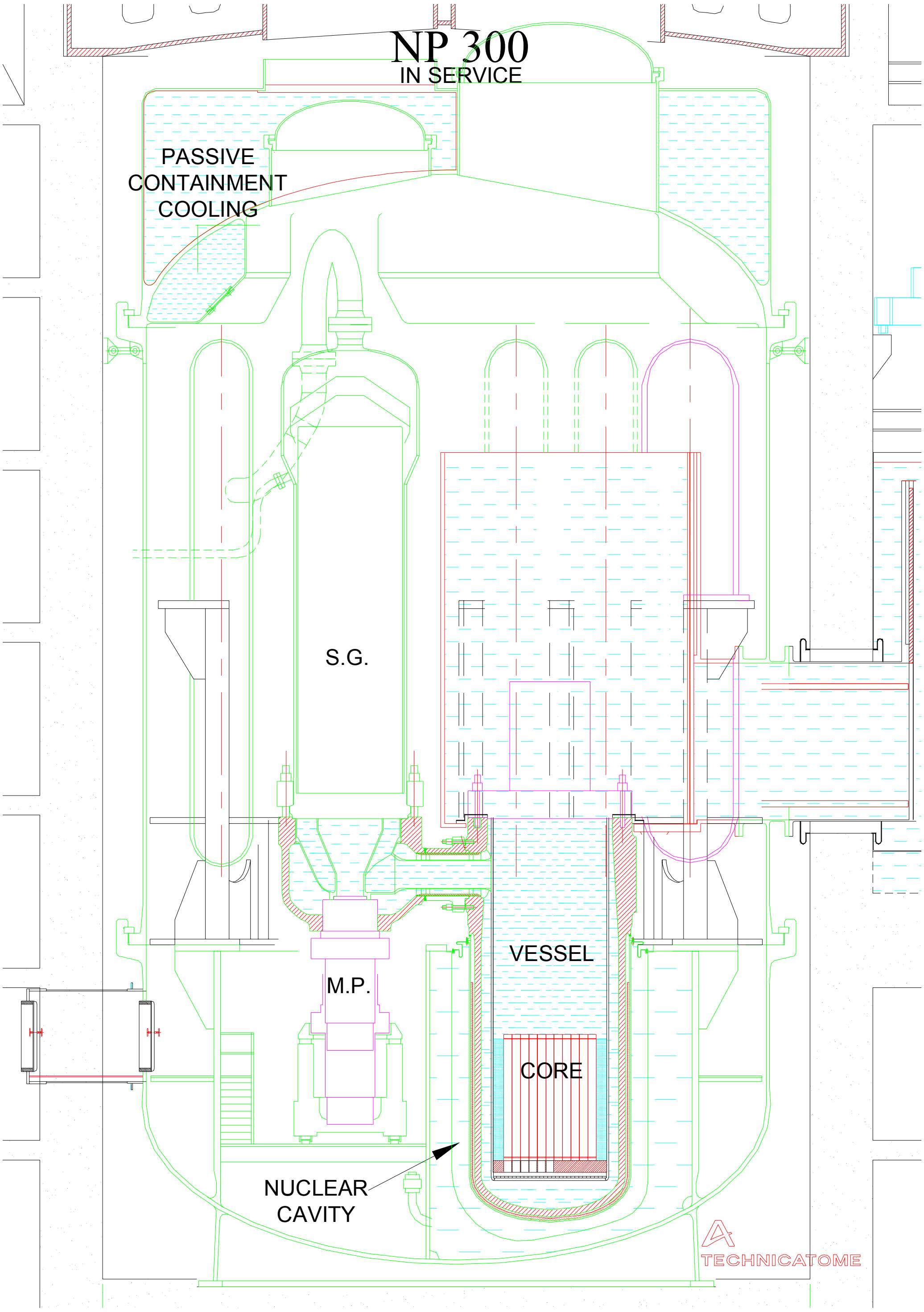


FIG. 5.10-4. NPP-300 concept.

5.10.6.2 Safety systems and features (active, passive and inherent)

Safety systems

Containment lake

A large volume of water, known as the containment lake, is maintained above the metal containment. This provides the very high thermal inertia required to provide cooling functions in the event of an accident.

The large volume of cold water in the containment lake, which can be easily replenished if necessary, is able to absorb without difficulty and solely through its inertial effect, most of the energy required for cooling the reactor and removing the residual power. The containment lake water storage system for reactor safety is entirely passive.

The existence of the "containment lake" is used to maximum effect in defining the passive safety systems, designed according to the type of accident considered. The "containment lake" is used for example to define all the passive systems backing up safety in the event of a serious core meltdown accident.

Upper water tank inside the containment (= Upper contained water tank)

In addition to the containment lake described above, which is outside the containment, there is also a large water tank inside the containment, in its top section, directly connected to the containment lake and passively cooled by it. This "upper contained water tank" is divided into several "independent sub-tanks" to provide the necessary segregation between different types of fluid.

Emergency cooling circuits ("contained" circuits) (2 lines at full capacity)

This system comprises two steam condensers contained in steam generators. They operate by natural circulation and are each controlled by motor-operated valves.

In principle these condensers are cooled by a pumped system during normal operation, but they are also designed to operate by natural circulation from the "containment lake" with the pumps shut down.

This system is organized on the basis of 2 independent lines at full capacity (1 condenser per steam generator). The system is designed in such a way that after reactor shutdown:

- The secondary pressure relief valves are not lifted under the following conditions:
  - initial primary temperature = nominal value,
  - steam isolation valves closed at initial moment,
  - only one condenser is in service,
  - startup time-delay = 10 mn after shutdown.

- Primary pressure drops to a value at which the safety injection tanks (SIS) can be activated, regardless of the type of depressurization accident (break).

It should be noted that the measures and dimensioning referred to above in relation to the letdown of the secondary pressure relief valves in the containment allow primary/secondary leakage events to be managed without risk of extensive contamination provided that the steam isolation valves are closed quite soon after detecting the event.

Safety injection systems (3 lines at half capacity) (Fig. 5.10-5)

The safety injection systems are arranged in 3 independent lines emerging directly in the vessel. Two of these lines are sufficient to provide the functions required.

Each system comprises, in the downstream to upstream direction from the vessel :

- A pressurized gas accumulator tank (nitrogen) installed in the (CCN)(*);
- A gravity-feed safety injection line (see below) installed in the CCN;
- A low-pressure injection and recirculation pump installed outside the CCN;
- A specific water tank installed outside the CCN.

The passive, gravity-feed safety injection line comprises a line connecting the contained water tank in the top section of the containment to the safety injection line.

The contained water tank is replenished passively during the post-accident period by a gravitational flow of condensed water from the steam condensers placed directly above it.

It should be noted that the measures described above make it physically impossible for water to be injected from the injection system (IS) into the vessel if the system is activated inadvertently.

Borated water is injected into the vessel via a pot containing boric acid pellets by means of a diaphragm tank filled with high pressure (HP) air.

(*) **C**ompartiment **C**haufferie **N**ucléaire

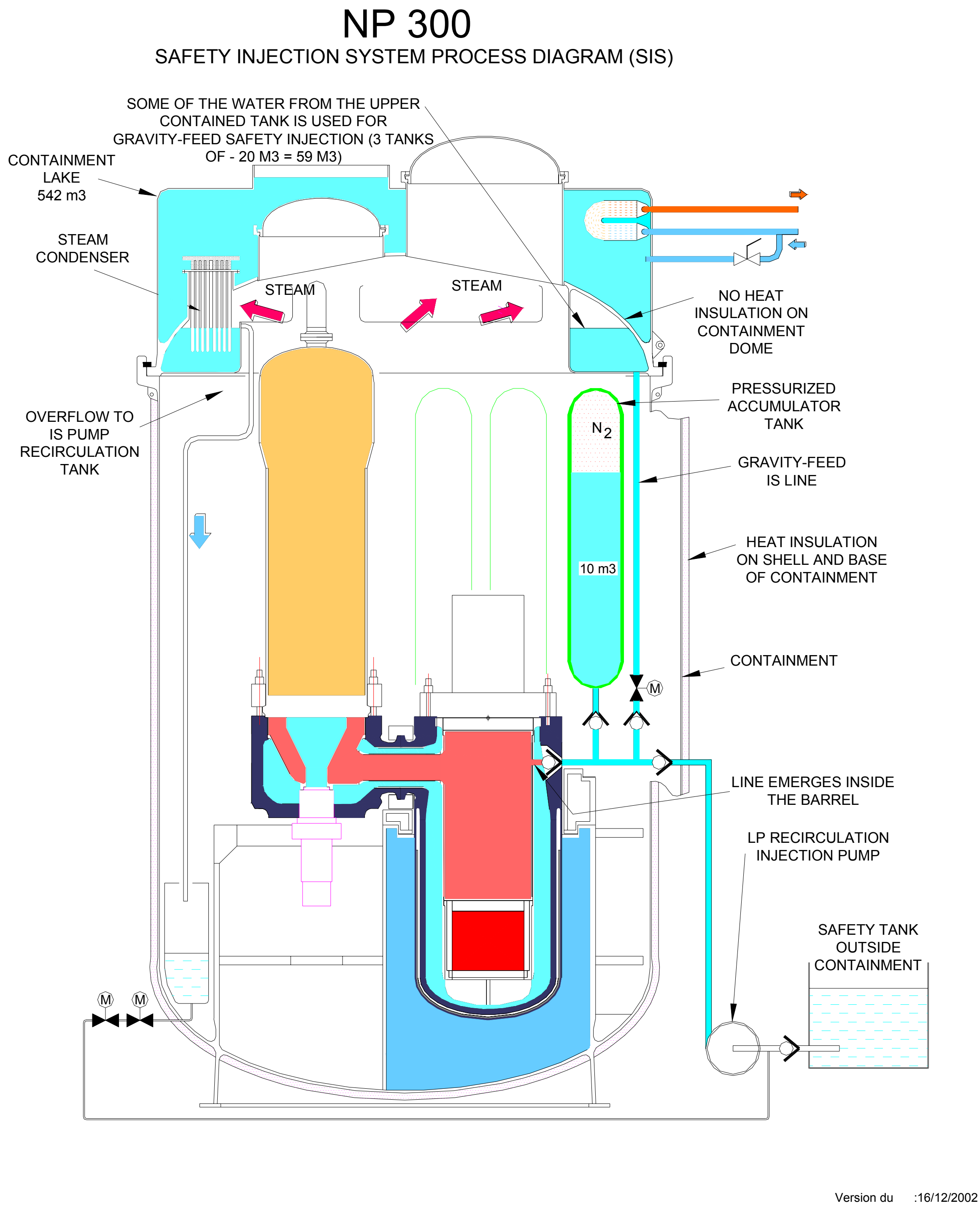


FIG. 5.10-5. Safety injection system process diagram.

Confining the corium to the vessel

In the hypothetical event of core meltdown, a system confines the corium in the vessel during the post-accident period.

Radiological protection

Radiological protection complies with updated CIPR 60 standards.

Non proliferation

The NP120/150 – NP-300 – NP450 reactor range uses proliferation resistant "sealed" fuel. Measures are taken to make IAEA safeguards inspection easier and ensure its reliability.

5.10.7 Plant layout

5.10.7.1 Buildings and structures, including plot plan (Figures 5.10-6(a) and (b))

All the strictly nuclear systems and equipment to be protected are contained in the reactor block buried below a protection slab.

The underground sections include:

- the reactor containment,
- the fuel pool,
- the control room and remote shutdown panel,
- the effluent rooms,
- the reactor electrical supply rooms (force and instrumentation & control),
- the safety systems.

At surface level:

- heavy-duty crane unloading area,
- offices,
- reception,
- locker rooms,
- ventilation.

The NP-300 reactor containment measures 28 meters in height by 16 meter in diameter. The concrete nuclear block is buried in a sealed, flood-protected excavation with sides of 40 meters and 35 meters deep.

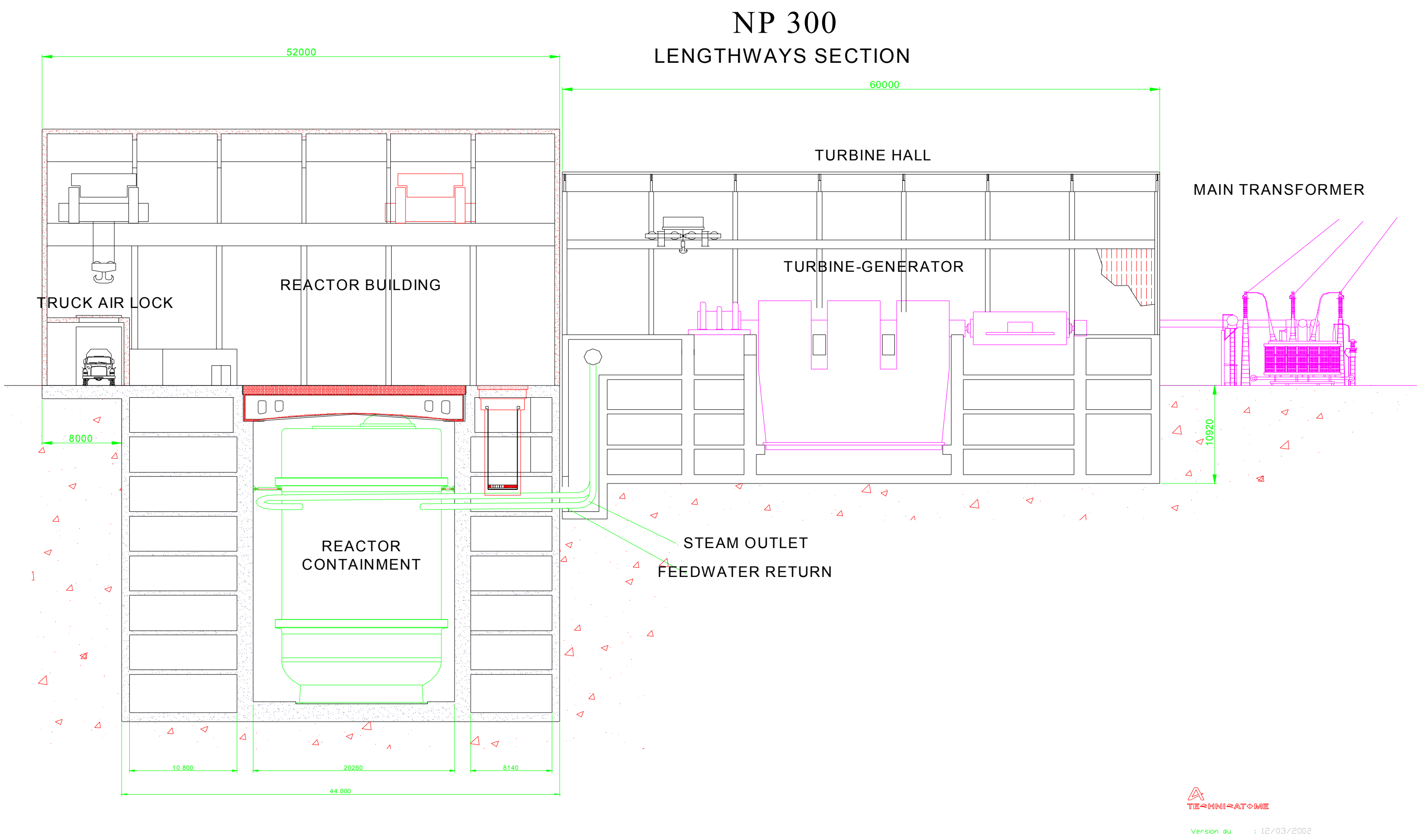


FIG. 5.10-6(a). Buildings and structures layout.

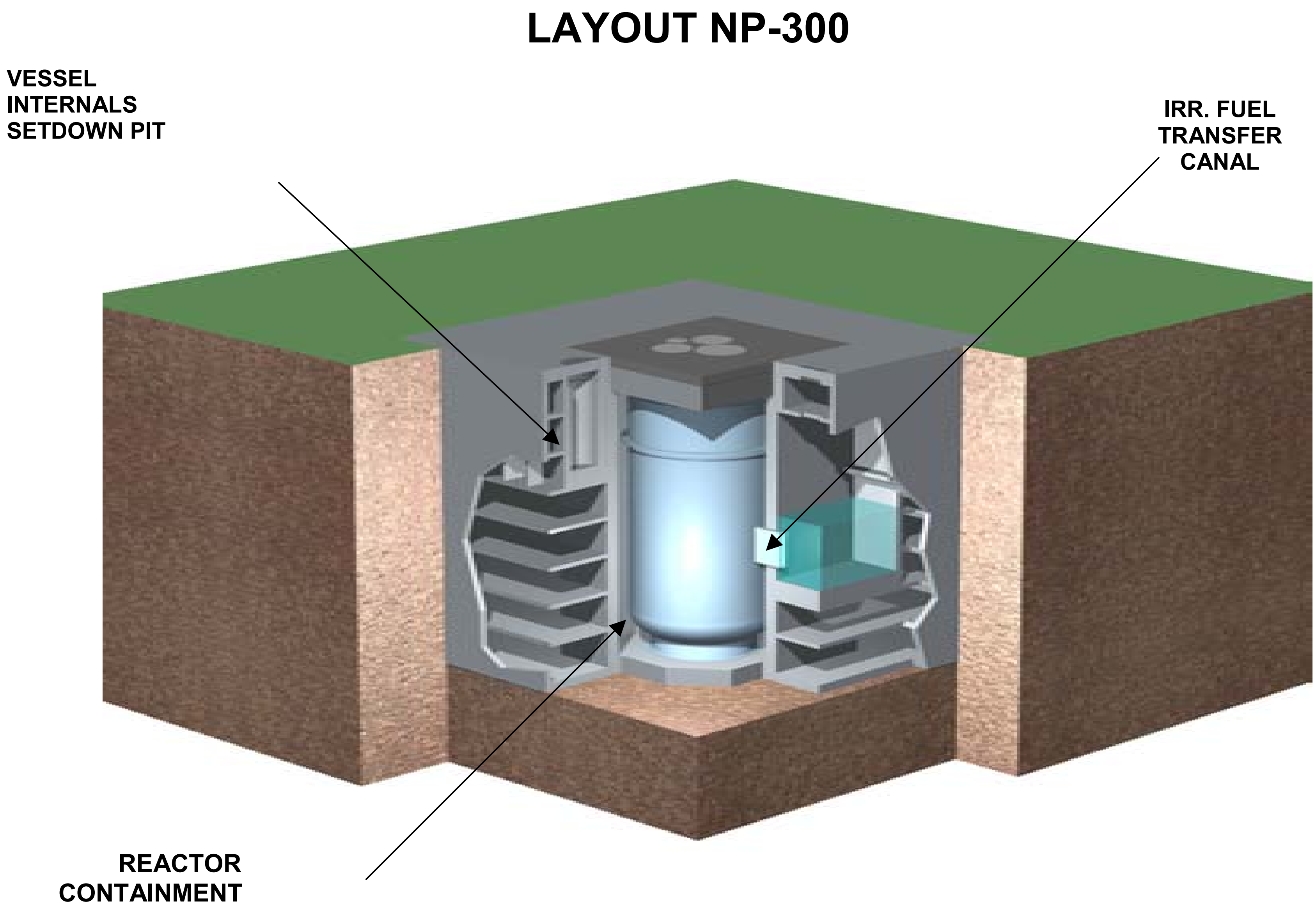


FIG. 5.10-6(b). Buildings and structures layout.

5.10.8 Technical data

Main Characteristics of the NP-300 (provisional values)
Nuclear reactor

Number of vessel-steam generator connectors	2
	coaxial circulation
Nominal power of the boiler plant	1000 MW(th)
Maximum permissible power of the boiler plant	1060 MW(th)
Nominal electrical power delivered by the generator	334 MW(e)
Net electrical power delivered to the grid	314 MW(e)
Type of fuel	17 x 17
Total length of assembly	3570 mm approx.
Number of assemblies	97
Weight of one assembly	536 kg approx.
Steam generator model	U-shaped tubes
Number of steam generators	2
Heat exchange area per steam generator	4100 m^2
Total height of a steam generator	15,5 m
Total weight (without water)	225 t
Steam output pressure	34.7 bar
Feedwater temperature	175 °C
Steam output rate (by 2 steam generators)	490 kg/s
Secondary design temperature (at steam generator)	305 °C
Secondary design pressure (at steam generator)	95 bar
Inner diameter of vessel at core level	3.3 m
Thickness of vessel shells	0.15 m
Total height between mating surface and inside of the bottom curve	8.61 m
Weight of vessel and steam generator channel heads	500 t

Type of primary pump	Submerged rotor
Number	2 (1 per GV)
Cold coupling power	2.5 MW(e)
Hot coupling power	2.2 MW(e)
Total hot pressure head	45 m WC

Nominal operating pressure of the primary system	100 bar
Primary system design pressure (hot)	130 bar
Primary design temperature (excluding pressurizer and LCP)	305 °C
Hydraulic operating data:	
– temperature at vessel outlet / steam generator inlet	287.9 °C
– temperature at steam generator outlet / primary pump inlet	254.1 °C
– core rate	5936 kg/s
– core heat exchange area	2500 m^2

Turbine-generator unit

<u>Turbine:</u>

– 1 single-flow MP module
– 2 double-flow LP modules

Rotation speed	1500 rpm
Architecture	1 MP cylinder and 2 LP cylinders
Overall length of turbine	
Last LP impeller diameter	
Number of bleedoffs	

<u>Turbine intake, MP level:</u>

flow rate	490 kg/s
– pressure	32.7 bar
temperature	238.6 °C

<u>HP outlet:</u>

pressure	5,8 bar

temperature	153 °C
<u>Dryer / superheater:</u>	
– number of superheat stages	
<u>LP intake:</u>	
flow rate	
– pressure	5.24 bar
temperature	221 °C
<u>Condenser pressure</u>	see note 1

<u>Generator:</u>		
nominal power	334 MW(e)	
power factor	voltage	rotor cooling
stator cooling	water	
rotor diameter		
rotor weight		
stator weight		

Note 1 : *If a river and cooling towers are used for cooling, the condenser pressure is 50 to 55 mbar; if seawater is used, the condenser pressure is 35 to 40 mbar.*

5.10.9 Measures to enhance economy and maintainability

THE NP-300 PLANT ADAPTS TO A WIDE VARIETY OF SITE CONDITIONS

To respond to the variety of sites at which the NP-300 must be able to operate and the particularly severe conditions of most of them, TECHNICATOME/AREVA has designed the NP-300 for easy adaptation (by simple adjustment of certain features) to the toughest possible constraints (severe external hazards, high earthquake levels, sandstorms, etc.) or special customer requirements.

THE NP-300 PLANT CAN BE INSTALLED ON A BARGE

The compact design NP-300 reactor, derived from technologies implemented in French nuclear propulsion systems, is also suitable for application as a floating power unit.

A barge-mounted floating NP-300 complex, modeled on offshore oil drilling platforms, has several financial, technical and operating advantages:

- Suitability for industrial-scale production (by assembling prefabricated process components) resulting in a significant saving in project costs;
- Towing to its installation site once assembled. It is installed in floating or anchored configuration or on underwater foundations, at short distances from the coast. This allows the corresponding coastal areas to be used for other purposes;
- Potential servicing of several geographical zones during its operating lifetime (50 years). At the end of this operating life, it can be towed to a dismantling site.

THE NP-300 IS WELL SUITED FOR DESALINATION OF SEAWATER

The NP-300 can develop 300 MW(e) for a thermal output of 1000 MW(th). If installed on the coastline, in addition to providing electricity, it can produce a daily volume of desalinated water (between 300,000 and 500,000 m^3/day) by using waste heat from the secondary system (700 MW(th)).

The plant can also be easily adapted to all types of desalination processes.

A demonstration unit combining a seawater desalination system with a reactor of the same type used at Cadarache is currently being installed.

This also includes three thermal distillation technologies:

- Multi Stage Flash (MSF),
- Multi Etaged Distillation (MED),
- Vapor Compression (VC).

and two membrane processes:

- Reverse Osmosis (RO),
- Electro Dialysis (ED).

5.10.10 Project status and planned schedule

The project schedule is as follows:

- Conceptual design ⇒ completed mid-2002
- Basic design ⇒ ready in June 2004
- Safety Analysis Report ⇒ January 2005
- Authorization and certification by the safety authority ⇒ June 2005
- Construction leadtime ⇒ 54 months

The experience base for the NP series includes 18 nuclear reactors that TECHNICATOME has developed and operated since 1964, representing at total of around 300 reactor-years of experience and operation.

TABLE 5.10-I EXPERIENCE BASE FOR THE NP SERIES IN FRANCE

Experience in France (reactor-years)			
	Nuclear reactors	**Year of startup**	**Reactor-years**
Test reactors	PAT	1964	26
	CAP	1975	15
	RNG	1989	13
Ballistic missile nuclear submarine, class Le Redoutable	Le Redoutable	1969	23
	Le Terrible	1971	26
	Le Foudroyant	1973	24
	L'Indomptable	1975	24
	Le Tonnant	1979	21
	L'Inflexible	1984	18
Nuclear attack submarine, class Rubis	Rubis	1981	20
	Saphir	1983	18
	Casablanca	1986	15
	Emeraude	1987	14
	Améthyste	1990	11
	Perle	1992	9
Ballistic missile nuclear submarine, class Le Triomphant	Le Triomphant	1995	6
	Le Téméraire	1998	2
Aircraft carrier	Charles de Gaulle	1998	3 x 2
Total			**291**
	PAT : Land-based prototype reactor CAP : Advanced propulsion reactor RNG : New generation reactor (renovation of the CAP)		

CHAPTER 6. SMALL SIZE ADVANCED LWR DESIGNS (BELOW 300 MW(e))

6.1 LSBWR (TOSHIBA CORP., JAPAN)

6.1.1 Introduction

In Japan, increases in nuclear plant unit capacity have been promoted to take advantage of economies of scale while further enhancing safety and reliability. As a result, more than 50 units of nuclear power plants are playing an important role in electric power generation. Currently, the next generation reactor with 1700 MWe is under development. However, the future of nuclear power generation looks uncertain because of increasing competition with other sources of power generation in the deregulated market in spite of the fact that nuclear power generation is generally recognized as an attractive option from the viewpoints of energy security and environment protection. Furthermore, the factors, such as stagnant growth in the recent electricity demand, limitation in electricity grid capacity and the desire to minimize risk by limiting the initial investment, will not be in favor of large plant outputs. Nuclear plants that can be easily adopted in any country are required in order to globalize nuclear power generation for greenhouse effect mitigation.

The reactor concept described in this section has a small power output, a capability of long operating cycle, and a simplified BWR configuration with comprehensive safety features. To be economically competitive, simplification of systems and structures, modularization for short construction period, and improvement in availability are included into the Long Operating Cycle Simplified BWR (LSBWR) design. For comprehensive safety features, the aim is to need no evacuation by utilizing highly reliable equipment and systems such as large RPV inventory, bottom located core layout, in-vessel retention (IVR) capability and passive Emergency Core Cooling System (ECCS) and Primary Containment Vessel (PCV) cooling.

The concept proposed here is to provide flexibility for various site conditions and electricity demands, to mitigate investment risks, and to facilitate public acceptance

The LSBWR design has the following objectives:

- Economically competitive with other source of power generation;
- Comprehensive safety feature with no evacuation.

To meet the power output targets, the LSBWR power range has been chosen from 100 to 300 MWe. To overcome the demerit by economies of scale with small size reactors, the LSBWR has taken into accounts the following technologies:

- Simplification of systems by combination of direct cycle, natural circulation and passive systems;
- Simplification of structures by integration of the reactor building and the turbine building into one;
- Elimination of the fuel pool and the refueling machines with super long operating cycle (in 15 years operation cycle case);
- Modular method for short construction period;
- Adoption of seismic isolation and ship hull structure;
- Improvement in availability by long operating cycle (3 to 15 years).

The LSBWR safety approach takes into account the currently available recommendations for future reactor concepts. For comprehensive safety, no evacuation is achieved by highly reliable equipment and systems as follows:

- Large RPV inventory,
- Bottom located core layout,
- IVR capability,
- Passive ECCS and PCV cooling.

6.1.2 Description of the nuclear systems

6.1.2.1 Primary circuit and its main characteristics

From the points of reactor simplification and severe accident countermeasures, the LSBWR reactor system is designed by employing highly reliable equipment and systems such as:

- Large RPV inventory,
- Bottom core configuration,
- IVR capability,
- Passive ECCS and PCV cooling.

Figure 6.1-1 shows the LSBWR reactor internals and configuration [Ref. 1]. Three distinctive features are as follows:

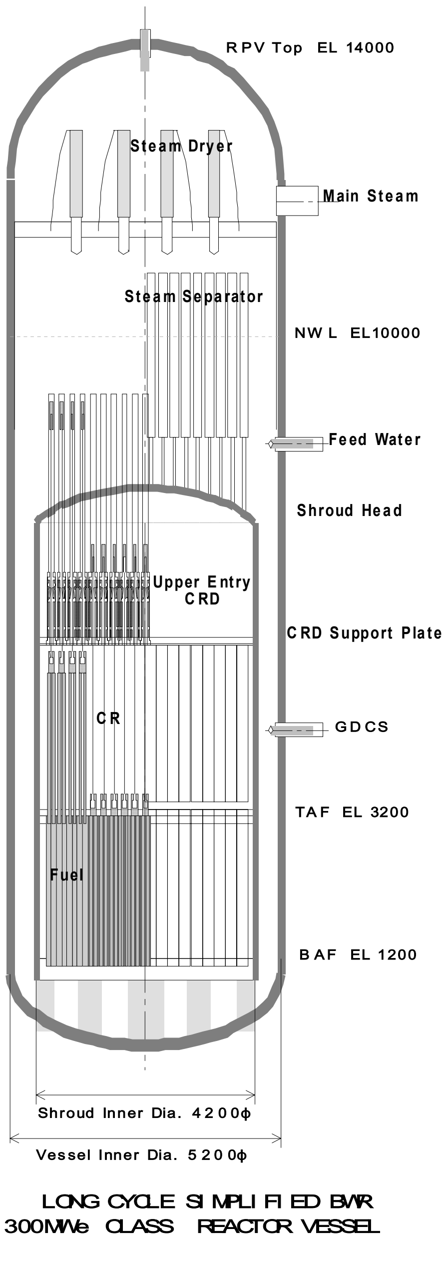


FIG. 6.1-1. LSBWR reactor concept

a. Natural circulation core cooling

Natural circulation core cooling is applied for eliminating recirculation pumps. This results in high reliability in operation. For attaining natural circulation core cooling, the fuel length is shortened to 2.2m from the conventional 3.7m length to decrease the pressure drop.

b. Internal upper entry control rod drive (CRD) [Ref. 2]

In BWR reactor design history, the lower entry CRD had been applied for over 30 years. It is difficult to design an upper entry CRD because of two-phase flow and separators and dryers. An innovative CRD has been under development as a configuration of internal mounted upper entry above the core in the LSBWR reactor. A guide chimney that has functions of CR guide and two-phase flow path above the core separates the path of the CR from the core flow, the CR is operated without flow-induced vibration (FIV).

c. Large RPV inventory

Internal upper entry CRD enables RPV bottom located core configuration. The natural circulation core and internal upper entry CRD configuration results in large water inventory above the core and a large safety margin against the loss of inventory.

6.1.2.2 Reactor core and fuel design

There are the following two stages for long cycle core development for LSBWR:

- The near term development: Using the conventional core design, long cycle operation is pursued;
- The long term development: Considering future deregulation and infrastructure changes, special core design for super long cycle core is developed.

a. Conventional design long cycle core

A conventional BWR design was further simplified to fit for a small reactor. The power density of the core was decreased and the fuel length was shortened. This decrease in power density results in simplification in the coolant circulation system of the BWR because the natural circulation is high enough for the core with such low power density and low pressure drop with shorter fuel. In general, BWRs have a high natural circulation capability in the RPV because of large difference in the coolant density between the inside and outside of the core shroud. The low power density core results in improvement not only in design simplification but also in availability of the plant. The low power density lengthens refueling intervals and consequently enhances availability of the plant. For example, a cycle length of three- effective-full-power-years (EFPYs) [Ref. 3] is achievable with the standard 45 GWd/t burnup of BWR fuels. Theoretically, the availability exceeds 95 % with the low power density core.

Major characteristics of the reactor are summarized in Section 6.1.8 together with a cross-sectional view of the reactor with the core configuration in Figure 6.1-1.

b. Long cycle core

One way to achieve the super long operating cycle (over the 15 years) is adoption of a high conversion core, which is attainable with a hard neutron spectrum [Ref. 4]. Some designs use a combination of a tight lattice core and plutonium MOX fuel to attain a hard spectrum [Ref. 5]. Instead of hardening the neutron spectrum, the LSBWR concept adopts the combination of medium enriched uranium oxide fuels and non-tight lattice bundle [Ref. 6], because this configuration facilitates natural circulation for core cooling. To realize this idea, the following challenges are addressed.

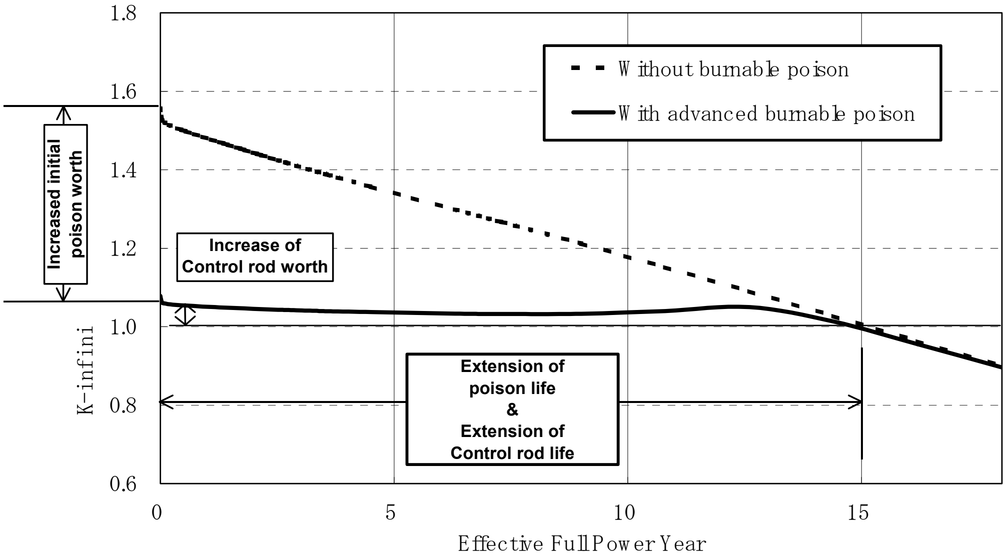


FIG. 6.1-2. Challenge to achieve super-long cycle operation

- Extension of reactivity life,
- Increase of control rod worth,
- Extension of control rod life.

Figure 6.1-2 shows the relationship among these challenges. Two lines indicate the k-infinity with or without burnable poison using a non-tight UO_2 lattice. Increased initial poison worth and extension of poison life are required for suppression of reactivity for a long cycle operation. The increase of control rod worth and extension of control rod life are also required to control the reactivity for super-long operation.

6.1.2.3 Fuel handling and transfer systems

The fuel handling and refueling operations are similar to those in current BWR design.

6.1.2.4 Primary components

Reactor pressure vessel

The reactor vessel is 19 m high with an internal diameter of 5.6 m. The operating pressure is 7MPa as in the ABWR. The nozzles for the two main steam lines of 500 mm diameter and the two feedwater lines of 250 mm diameter are located in the upper portion of the vessel. Though a conventional BWR's reactor pressure vessel has the CRD penetrations at the bottom, there are no CRD penetrations at bottom of the LSBWR reactor vessel because internal upper entry CRDs are applied.

Reactor internals

In BWR reactor design history, lower entry CRDs had been applied over 30 yeas. It is difficult to design upper entry CRDs because of the two-phase flow and the steam separators and dryers. Therefore, an innovative CRD has been developed as a configuration of internal mounted upper entry above the core in the reactor. As guide chimney that have functions of CR guide and two-phase flow path above the core is separating the path of CR from core flow, CR is operated without flow-induced vibration (FIV). For escaping an interference of CRDs and separators, offset square layout design [Ref. 7] have been considered between CRDs and separators as shown in Figure 6.1-3.

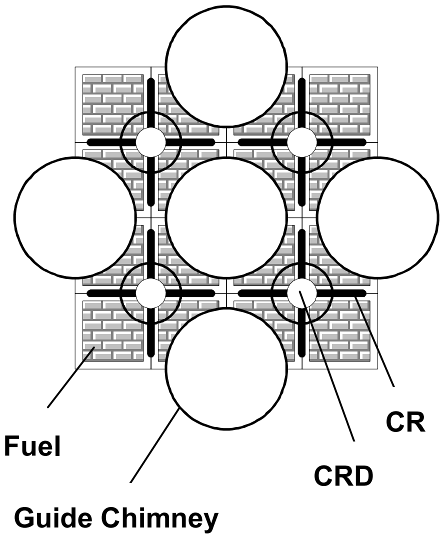


FIG. 6.1-3. Layout co-relation of CRD and chimney

CRDs

An innovative CRD has been developed as configuration of internal mounted upper entry above the core in the reactor as shown in Figure 6.1-3. Internal upper entry CRD is based on electromagnet coupling and motor driven technologies. Internal upper entry CRD consists of a CR driving motor, latch mechanism for scram, position indicator and electromagnet coupling. Electromagnet coupling is operated to transfer signal and electric power between outside and inside of RPV without contact. All these electric coils are shielded by ceramics that could withstand to high temperature. This innovative internal upper entry CRD are being developed under a government sponsored program.

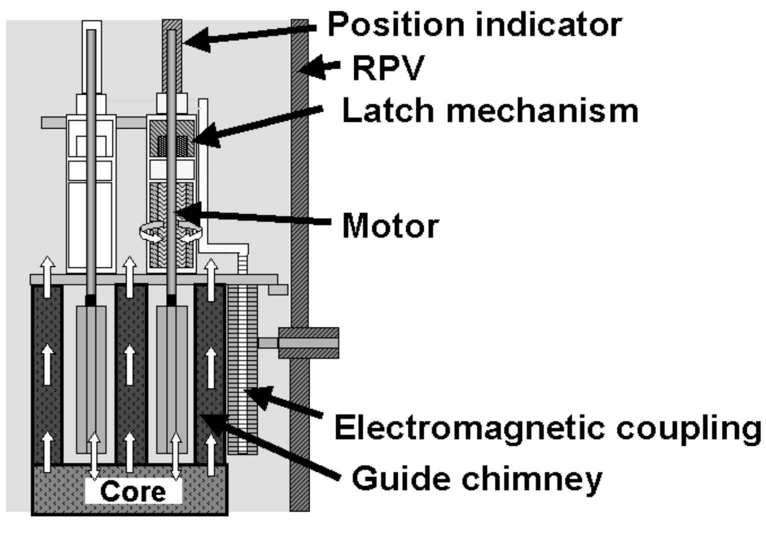


FIG. 6.1-3. Internal upper entry CRD

Reactor recirculation pumps

The LSBWR is designed with natural circulation of the coolant and there are no recirculation pumps.

6.1.2.5 Reactor auxiliary systems

A reactor core isolation cooling (RCIC) system is installed to maintain the reactor water level by water injection from the suppression pool during isolation events. A reactor water clean-up (CUW) system is installed to control the content of radioactive and corrosion products in the reactor water. A residual heat removal (RHR) system is installed to remove residual heat during reactor shutdown and to cool the primary containment spray at DBA events.

Other auxiliary systems such as the spent fuel storage pool cooling are primarily same as those for current BWRs.

6.1.2.6 Operating characteristics

LSBWR is being designed to improve in availability by long operating cycle (2 to 15 years).

6.1.3 Description of the turbine generator plant systems

The balance of plant of the LSBWR is mainly composed of two main steam lines, two feedwater lines, including one high pressure turbine and two low pressure turbines, etc. The main turbine is a tandem compound, double flow turbine with 42" blades on the last stage of the low pressure (LP) turbine units.

6.1.4 Instrumentation and control systems

The concept of I&C and electrical system in LSBWR are basically simplified from those of the ABWR. These simplifications are achieved by simplified system itself, due to adoption of passive safety systems and elimination of recirculation systems.

6.1.5 Electrical systems

Electrical systems for LSBWR are being designed based on current BWRs. The power supply system capacity is greatly reduced due to simplified system, adoption of passive safety systems and elimination of recirculation systems.

6.1.6 Safety concept

6.1.6.1 Safety requirements and design philosophy

The safety design philosophy is as follows:

- To increase reliability of systems and avoid system complexity by adding passive safety systems;
- Comprehensive safety feature with no evacuation.

The LSBWR safety approach takes into account the currently available recommendations for future reactor concepts. For comprehensive safety, no evacuation is achieved by highly reliable equipment and systems as follows:

- Large RPV inventory,
- Bottom located core layout,
- In Vessel Retention (IVR) capability,
- Passive ECCS and PCV cooling.

6.1.6.2 Safety systems and features (active, passive and inherent)

LSBWR safety system concept is shown in Figure 6.1-4.

The cylindrical type drywell with small diameter can be designed by routing the safety relief valve piping through the spacing between the RPV and the drywell wall and the main vent pipe from the RPV top to the suppression pool. The drywell air space is minimized and contains only SRV and depressurization valve (DPV) components, the gravity driven core cooling system (GDCS) and drywell loading piping. Since MS and FW piping is routed through suppression pool air space, which are protected by the guard pipe, GDCS piping is contained in the access tunnel placed in the lower part of suppression pool, and isolation valves are installed outside PCV etc.

Since the reactor core is placed at the bottom of the RPV, the emergency coolant injection system consisted of DPV and GDCS can achieve high reliability of the water coverage of the reactor core following an accident.

The containment wall with ship hull structure [Ref. 8] is filled with cooling water that is boiled off to the atmosphere to cool the PCV passively during an accident. This containment wall cooling system is also used for the drywell cooling during the normal operation and therefore the drywell arrangement is simplified without drywell cooling component used in the current BWR containment.

When cooling water in the PCCS pool above PCV is exhausted, external pool or seawater is supplied by gravitation so that the highly reliable and long term PCV cooling is achieved.

The double cylindrical raised suppression pool with the ship hull structure is installed around the cylindrical drywell and above the core elevation. This results in the simpler and stronger structure, and the suppression pool water can be easily used for GDCS and drywell lower part flooding.

As to the system dealing with design basis accidents (DBA), a PCV spray cooling system using active residual heat removal system (RHR) and active stand-by gas treatment system (SGTS) is used in addition to the GDCS and the PCV wall cooling mentioned above. The PCV pressure following an accident can be decreased to near atmospheric pressure; then the radioactive release to the environment is minimized. The RHR has low-pressure core flooder function as a backup. Two system trains are enough for this configuration assuming a single failure, and two small diesel generators (DGs (or gas-turbine generator (GT)) are installed.

Passive flammable gas control system (FCS) is installed in the suppression pool air space and vent tank space.

The emergency condenser is installed in the PCCS pool. The reactor core isolation cooling system (RCIC) is also installed for makeup during a transient and a small piping break accident similar to the current BWR. The water source in the main turbine hot well is initially used and the suppression pool is finally used.

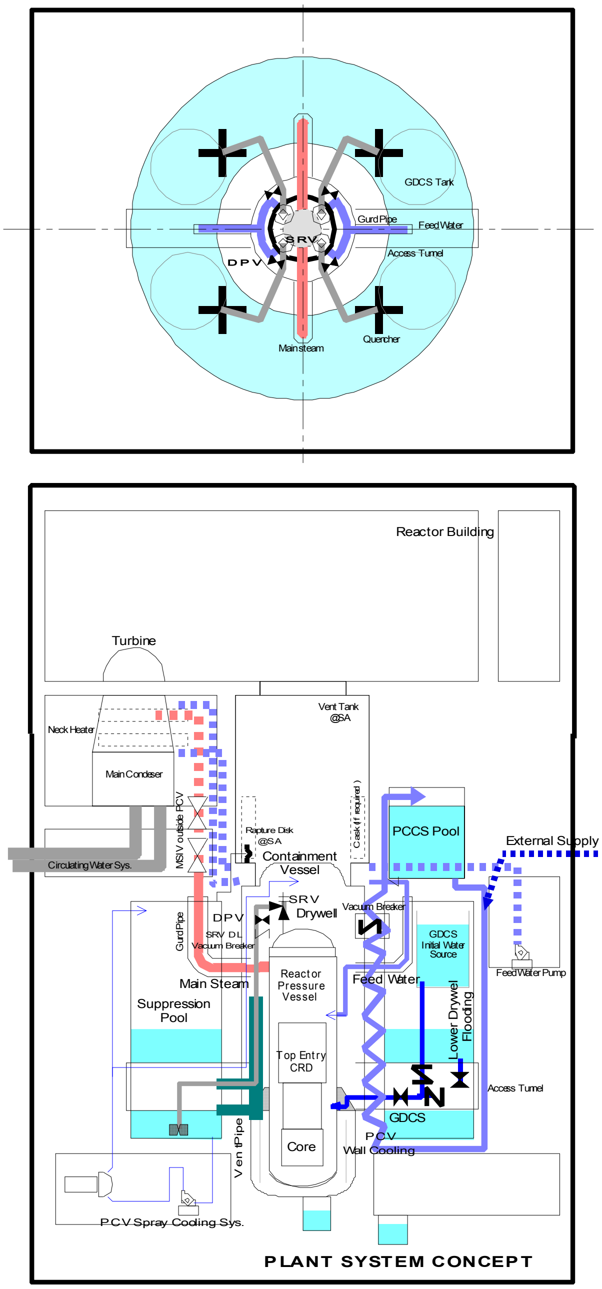


FIG. 6.1-4 LSBWR safety system concept

6.1.6.3 Severe accidents

There is open space above the PCV, which is called "reactor well" in a current BWR. In the LSBWR, this portion is used for vent tank space during a severe accident. By using the vent tank space the suppression pool air space can be rationalized and the no containment vent to the environment in case of a severe accident is required. In addition, the PCV pressure can be decreased by depleting the flammable gas using the passive FCS with a catalyst. This Passive FCS (so-called A-PAR) has additional function to convert Hydrogen to ammonia NH_3 with Nitrogen in the PCV [Ref. 9].

Since the CRD housing tubes are removed from the RPV bottom, the RPV outer wall can be easily cooled by flooding the lower-drywell with the suppression pool water in case of a so-called severe accident where a core melt is assumed. Even in severe accidents, the molten core is cooled and maintained inside the RPV through RPV bottom cooling. This performance is known as in vessel retention (IVR).

6.1.7 Plant layout

For achieving a short construction period and high production quality, module fabrication and construction are studied for LSBWR. In our system, module means not only system equipment, but also building structure. As LSBWR is small size scale plant, it is possible to fabricate, transport and construct in one-piece as a whole plant if transport conditions to the site allow. So far, module construction methods were generally applied to some components assemblies such as pump and its supporting equipment. Since the reactor building was usually a reinforced concrete structure, it was impossible to fabricate component module with building module. In shipbuilding industry, ship hull structure is applied for large size ship building such as 500,000 tons class. Though ship hull structure is lighter than reinforced concrete structure, it has enough strength and appropriate characteristics to apply for the nuclear reactor building. By using this ship hull structure, it is possible to fabricate modules containing component and a part of the building at a shop.

Shipbuilding technology employs advanced automation, and remarkably improved assembly lines are used as a result of competing in a severe international market. The hull structure of a large ship is constructed simply with steel plates. The basic structure of the hull consists of steel plate, girder (large beam) and stiffener (small beam) as shown in Figure 5. Almost the entire process, including the receipt of materials, forming large blocks, the welding process, and the removal of distortion after welding, is performed automatically on an assembly line. The large blocks are transferred to a shipbuilding dock where they are assembled to shape a hull. When most of the construction work is completed, the hull is launched and adjusted.

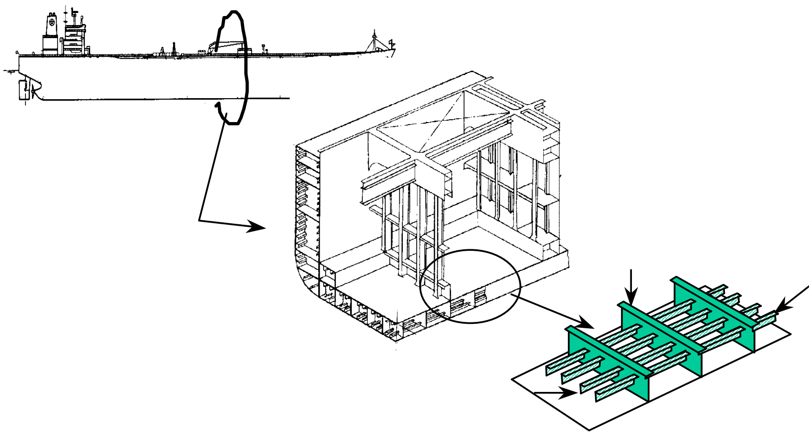


FIG. 6.1-5. Basic structure of hull structure

Ordinary ships have the 'single' steel plate lattice structure. But the 'double' steel plate lattice structure, two steel plates sandwiched together, is also being used for the latest tankers to prevent oil leaks. Ordinary walls and floors of a reactor building can be designed using the single steel plate lattice structure. For places where radiation shielding is required, concrete is filled up into the double steel plate lattice structure as one of the various methods. Since supporting brackets for piping and foundations for machines can be welded to walls or floors directly, the installation work will be simple compared with that in conventional buildings.

In the LSBWR building design, the reactor building and the turbine building are combined into one. Because the LSBWR is small and lighter than medium or larger size plants, it is possible to mount the turbine system on the upper part of the reactor building. One building arrangement would reduce building volume and base isolation structure can be used. By applying the base isolation structure to whole building, it becomes possible to make building module standardized regardless of various site seismic conditions. General arrangement of LSBWR is shown in Figure 6.1-6. As LSBWR has GDCS and PCCS, a raised suppression pool, GDCS pool and PCCS pool are located upper part of reactor core.

This unique building concept results in remarkable building cost reduction from building volume reduction and standardized shop fabrication.

Taking into account the characteristics of LSBWR concept, various kinds of construction methods are studied. From ABWR construction experience, it took 48-months (rock inspection to commercial operation) for standard ABWR construction period with concrete structure building. This 48-months construction period is shortened to 35.5 months by applying ship hull structure building**.** Construction period means construction works at the site in this section. The construction period of the LSBWR has been estimated 20 to 30 months with hull structure building. This construction period shows remarkable reduction compared with conventional construction period.

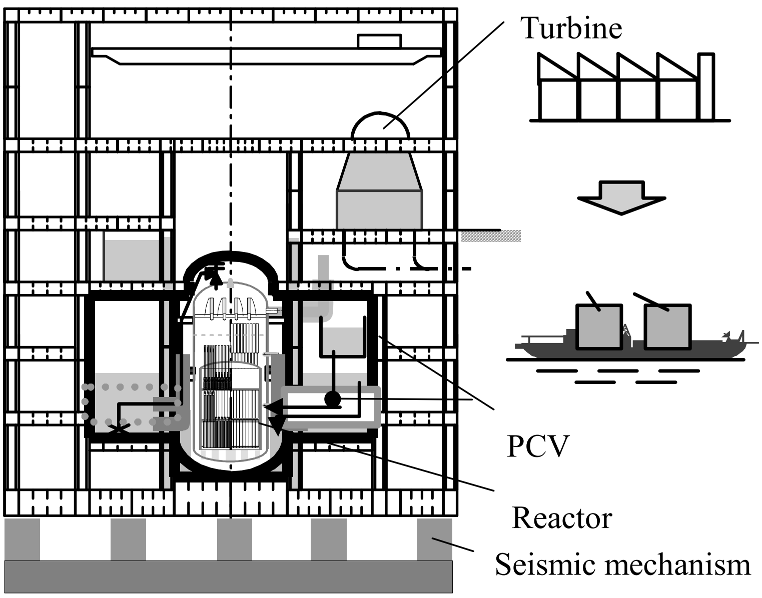


FIG. 6.1-6. LSBWR building concept

6.1.8 **Technical data** (all tentative)

General plant data

Power plant output, gross	306	MWe
Power plant output, net		MWe
Reactor thermal output	900	MWth
Power plant efficiency, net	34	%
Cooling water temperature	28.0	ºC

Nuclear steam supply system

Number of coolant loops	1	
Primary circuit volume		m^3
Steam flow rate at nominal conditions	486	kg/s
Feedwater flow rate at nominal conditions	486	kg/s

Reactor coolant system

Primary coolant flow rate	2750	kg/s
Reactor operating pressure	7	MPa
Steam temperature/pressure	286/7	^{o}C/MPa
Feedwater temperature	216	^{o}C
Core coolant inlet temperature	276	^{o}C
Core coolant outlet temperature	286	^{o}C
Mean temperature rise across core	10	^{o}C

Reactor core

Active core height	2.2	m
Equivalent core diameter	3.5	m
Heat transfer surface in the core		m^2
Average linear heat rate		kW/m
Fuel weight	42	tU
Average fuel power density	21	kW/kg U
Average core power density	42	kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		kJ/kg
Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	2900	mm
Rod array	(12X12), square lattice	
Number of fuel assemblies	284	
Number of fuel rods/assembly	128	
Number of spacers		
Enrichment (range) of first core, average	5 to (18)*	Wt%
Enrichment of reload fuel at equilibrium core	5	Wt%
Operating cycle length (fuel cycle length)	24 to (180)*	months
Average discharge burnup of fuel [capability]	55 to (110)*	GWd/t
Cladding tube material	zircaloy	
Cladding tube wall thickness		mm
Outer diameter of fuel rods	10.3	mm
Fuel channel/box; material	zircaloy	
Overall weight of assembly, including box		kg
Uranium weight/assembly	148	kg
Active length of fuel rods	2.2	m
Burnable absorber, strategy/material	axial and radial grading / Gd_2O_3 mixed with fuel	
Number of control rods	69	
Absorber material	B_4C and Hafnium	
Drive mechanism	motor driven	
Positioning rate		mm/s
Soluble neutron absorber	Boron	

Reactor pressure vessel

Inner diameter of cylindrical shell	5600	mm
Wall thickness of cylindrical shell		mm
Total height, inside	19000	mm
Base material: cylindrical shell	SFVQ1A (ASTM A508)	
RPV head	SFVQ1A (ASTM A508)	
lining	Stainless steel	

Design pressure/temperature		8.7/302	MPa/°C
Transport weight (lower part w/rigging)			t
RPV head			t

Reactor recirculation pump

Type	None	
Number	None	
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 Steel plate	
Overall form (spherical/cyl.)	Cylindrical	
Dimensions (diameter/height)		m
Separated Drywell		m
Design pressure/temperature		kPa/°C
Design leakage rate		vol%/day
Is secondary containment provided?	Yes	

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure (100°C)		kg/s
Coolant injection,	at high pressure (HPCF)	None	kg/s
	(ARCIC)		kg/s
	at low pressure (LPFL)		kg/s

Power supply systems

Main transformer,	rated voltage		kV
	rated capacity		MVA
Plant transformers,	rated voltage (secondary)		kV
	rated capacity		MVA
Start-up transformer	rated voltage (secondary)		kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)			kV
Number of low voltage busbar systems			
Standby diesel generating units;	number	1	
	rated power		MW
Standby gas turbine generating units;	number	1	
	rated power		MW
Number of diesel-backed busbar systems			
Voltage level of these			kV AC
Number of gas turbine-backed busbar systems			
Voltage level of these			kV AC
Number of DC distributions			
Voltage level of these			V DC
Number of battery-backed busbar systems			
Voltage level of these			V DC

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	Dual flow, tandem compound	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/1 LP/2 ,	
Turbine speed	3000	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	7/280	MPa/°C

Generator

Type	3-phase, turbo-generator	
Rated power		MVA
Active power	300	MW
Voltage		kV

Frequency	50	Hz
Total generator mass, including exciter		t
Overall length of generator		m

Condenser

Type	shell type (2 shells)	
Number of tubes	2 tube bundles/shell	
Heat transfer area		m^2
Cooling water flow rate		m^3/s
Cooling water temperature		$^{\circ}C$
Condenser pressure (HP shell)	5	kPa

Condensate pumps

Number		
Flow rate		kg/s
Pump head		MPa
Temperature	34	$^{\circ}C$
Pump speed		rpm

Condensate clean-up system

Full flow/part flow		
Filter type (deep bed or rod type)		

Feedwater tank

Volume		m^3
Pressure/temperature		$MPa/^{\circ}C$

Feedwater pumps

Number		
Flow rate	486	kg/s
Pump head	7	MPa
Feed pump power		MW
Feedwater temperature (final)	216	$^{\circ}C$
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages,	low pressure	
	high pressure	
	feedwater tank	1 (deaerator)

*Note: values relevant to the super long operating cycle of 180 months are designated by (__)**

6.1.9 Measures to enhance economy and maintainability

6.1.9.1 Simplifying the design

The LSBWR is designed to simplify systems, components and building as follows.

- Reactor system

  a. Natural circulation core cooling:
  Natural circulation core cooling is applied for eliminating recirculation pumps.
  b. Internal upper entry CRD:
  Internal upper entry CRD could simplify the bottom portion of RPV because of eliminating CRD guide tubes. Internal upper entry CRD could also eliminate CRD maintenance space below the bottom of RPV.

- Safety system

  Following passive systems could simplify the safety systems:
  a. GDCS:
  GDCS eliminates the active components such as high and low pressure injection pumps.
  b. PCCS:
  PCCS eliminates the active components such as pumps.
  c. A-PAR
  PAR+ eliminates the active components such as pumps.

- Building
  a. Integration of the building:
  Integration of the reactor building and the turbine building into one simplify the building structure and reduce the building volume. And integration of the building could apply the base isolation structure.
  b. Ship hull structure
  LSBWR building is designed to apply ship hull structure. Ship hull structure building is constructed simply with steel plates. This structure could make possible easy module fabrication at a shop.

6.1.9.2 Improvement of the operation flexibility

For improving the operation flexibility, LSBWR can apply a long operating cycle core. From the point of regulation change and infrastructure of fuel fabrication and handling, the following two staged approach is being studied for long cycle core development:

- The near term development: Using the conventional core design, long cycle operation is pursued.
- The long term development: Considering future deregulation and infrastructure changes, special core design for super long cycle core is being developed (option)

Targeted operating cycle length by the conventional core design of the LSBWR core is 2 to 7 effective full power years (EFPY). The other special core design for super long cycle is 15 EFPY.

Passive safety systems are applied to LSBWR design. These systems cover both DBA and SA condition, and practically eliminate the necessity of containment venting before and after core damage as a means of overpressure protection. Flammable gas control in the containment is performed by the combination of inerting and passive special autocatalytic recombiner (PAR+) which has advantages both of safety (automatic startup and passive operation) and economy (low cost, flexible layout and easy maintenance).

6.1.9.3 Reduction of the cost of equipment and structures

To overcome demerit by economies of scale with small size reactors, the design has taken into account the following technologies:

- Simplification of systems by combination of direct cycle, natural circulation and hybrid (Active+Passive) safety systems
- Simplification of structures by applying ship hull structure for building.
- Modular fabrication method for short construction period
- Integration of the reactor building and the turbine building into one simplify the building structure and reduce the building volume.

Building of LSBWR specific volume (m3/kWe) and specific weight (ton/kWe) are shown in Figure 6.1.7. From these results, building of LSBWR specific volume has become approximately 3/4 that of the ABWR, and specific weight has become approximately 60% of that of the ABWR. These results shows effects of innovative concept such as one-piece building arrange, ship hull structure building and system simplification. It is expected that these result in effective cost reduction in spite of demerits by economics of scale.

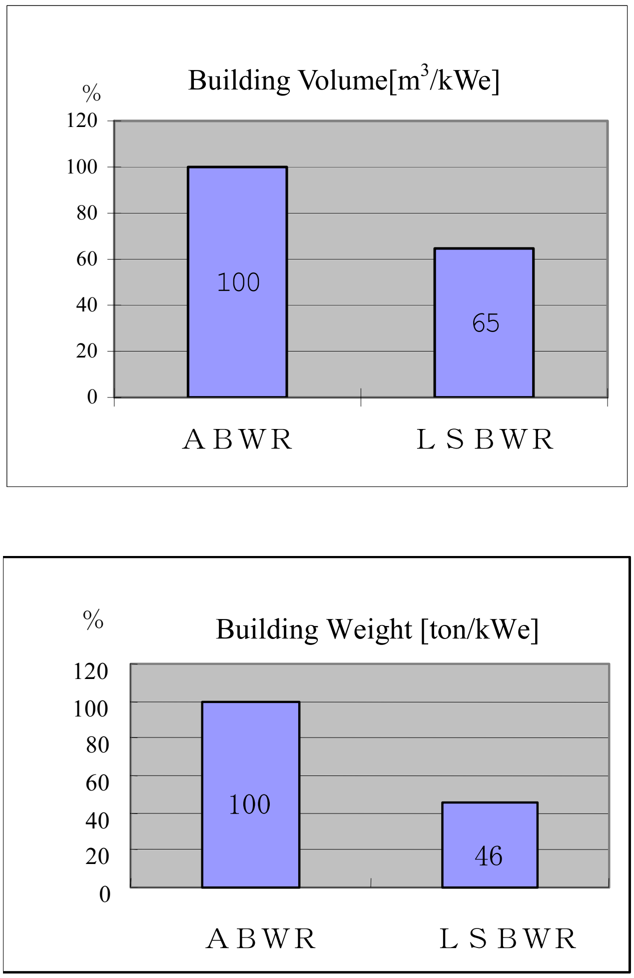


FIG. 6.1.7 Specific volume and weight comparison with ABWR building

6.1.9.4 Reduction of the construction period

Taking into account the characteristics of the LSBWR concept, various kinds of construction methods are studied. From ABWR construction experience, it took 48-months (rock inspection to commercial operation) for standard ABWR construction period with concrete structure building. This 48-months construction period is shortened to 35.5 months by applying ship hull structure building [3]. Construction period means construction works at the site in this paper. Construction period of LSBWR has been estimated 20 to 30 months with hull structure building. This construction period shows remarkable reduction compared with conventional construction period.

Each module block is designed within 1000 tons for easy handling by crane capacity of 1000 tons. This fabrication procedure will be applied to LSBWR. The shortest construction period is attained by one-piece construction method. In one-piece construction method, the whole plant in one piece is transported to the site by barge. This method is suited for coast side, big river side, and lake side that have possible approach routes by barge. Divided module construction period is estimated approximately 30 month. The construction periods mentioned above are influenced by fuel loading timing and system inspection timing.

6.1.9.5 Reduction of the scope of maintenance during operation and outages

The large 1.2 C-lattice core arrangement enables to decrease numbers of fuel bundles and CRDs. These contribute to additional refueling time reduction and CRDs maintenance time reduction.

Passive systems such as PCCS and PAR are applied for LSBWR. Passive system needs not to do maintenance work as active system. Therefore, scope reduction of the maintenance during operation and outages is achieved against passive related system.

6.1.9.6 Easier maintenance and with lower radiation exposure

The large 1.2 C-lattice core arrangement enables to decrease numbers of fuel bundles and CRDs. These contribute to additional refueling time reduction and CRDs maintenance time reduction. Therefore the large 1.2 C-lattice core arrangements make the maintenance easier and with lower radiation exposure.

Passive system such as PCCS and PAR+ are applied for LSBWR. Passive system needs not to do maintenance work as active system. Therefore, easier maintenance with low radiation exposure is achieved against passive related system.

6.1.9.7 Increasing the plant availability and load factor

Long operating cycle and refueling outage period reduction is a major factor in the plant availability and load factor for LSBWR.

Examples of design for maintenance reduction for increasing the plant availability and load factor are:
- Reduced number of fuel bundles and CRD
- Elimination of active components such as pumps by applying passive system.

6.1.9.8 Reduction of the power generation cost

The power generation cost is largely divided into three categories: capital cost, operation and maintenance (O & M) costs and fuel cycle cost. Simplification of the system decrease the cost both capital cost and O&M cost. Various design simplifications are applied to LSBWR such as natural circulation, passive systems and integration of the building. Fabrication and construction process simplification is also applied to LSBWR as module fabrication using ship hull structure. Capital cost reduction by standardization and/or mass production is attained by module fabrication and seismic isolation. Long operating cycle and refueling outage period reduction is a major factor in the reduction of O&M cost.

6.1.10 Project status and planned schedule

The LSBWR design is still at the conceptual design stage and licensing reviews have not yet been started.

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6.2 CAREM (CNEA/INVAP, ARGENTINA)

6.2.1 Introduction

CAREM is an Argentine project to achieve the development, design and construction of an innovative, simple and small Nuclear Power Plant (NPP). This nuclear plant has an indirect cycle reactor with some distinctive and characteristic features that greatly simplify the design, and also contributes to a higher safety level. Some of the high level design characteristics of the plant are: integrated primary cooling system, self-pressurised primary system and safety systems relying on passive features.

The CAREM concept was first presented in March 1984 in Lima, Peru, during the IAEA conference on small and medium size reactors. CAREM was, chronologically, one of the first of the present new generation of reactor designs. The first step of this project is the construction of the prototype of about 27 MW(e) (CAREM-25). This project allows Argentina to sustain activities in the nuclear power plant design area, assuring the availability of updated technology in the mid-term [1]. The design basis is supported by the cumulative experience acquired in Research Reactors design, construction and operation, and Pressurized Heavy Water Reactors (PHWR) Nuclear Power Plants operation as well as the development of advanced design solutions [2].

CAREM has been recognized as an International Near Term Deployment (INTD) reactor by the Generation IV International Forum (GIF).

6.2.2 Description of the nuclear systems

CAREM is an indirect cycle reactor with some distinctive features that greatly simplify the design and also contributes to a high safety level. Some of the high level design characteristics are:

- Integrated primary cooling system;
- Self-pressurised;
- Safety systems relying on passive features;
- Balanced and optimized design with a cost-effective internalization of safety.

6.2.2.1 Primary circuit and its main characteristics

The CAREM nuclear power plant design is based on a light water integrated reactor. The whole high-energy primary system, core, steam generators, primary coolant and steam dome, is contained inside a single pressure vessel (Fig. 6.2-1).

For low power modules (below 150 MW(e)), the flow rate in the reactor primary systems is achieved by natural circulation. Figure 6.2-1 shows a diagram of the natural circulation of the coolant in the primary system. Water enters the core from the lower plenum. After been heated the coolant exits the core and flows up through the riser to the upper dome. In the upper part, water leaves the riser through lateral windows to the external region. Then it flows down through modular steam generators, decreasing its enthalpy. Finally, the coolant exits the steam generators and flows down through the down-comer to the lower plenum, closing the circuit. The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form losses, producing the adequate flow rate in the core in order to have the sufficient thermal margin to critical phenomena. Reactor coolant natural circulation is produced by the location of the steam generators above the core. Coolant acts also as neutron moderator.

For high power modules (over 150 MW(e)) pumps are used to achieve the flow rate needed to operate at full power.

Self-pressurization of the primary system in the steam dome is the result of the liquid-vapor equilibrium. The large volume of the integral pressuriser also contributes to the damping of eventual pressure perturbations. Due to self-pressurisation, bulk temperature at core outlet corresponds to saturation temperature at primary pressure. Heaters and sprinkles typical of conventional PWRs are thus eliminated.

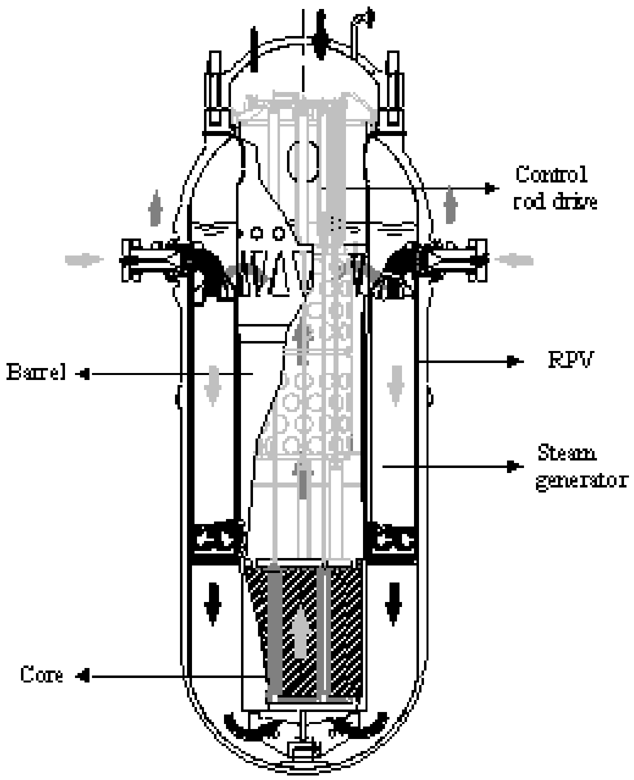


FIG. 6.2-1. Reactor pressure vessel.

6.2.2.2 Reactor core and fuel design

The Core has Fuel Assemblies (FA) of hexagonal cross section. Each fuel assembly contains 108 fuel rods of 9mm outer diameter, 18 guide thimbles and 1 instrumentation thimble (Fig. 6.2-2). Its components are typical of the PWR fuel assemblies. The fuel is enriched UO_2. Core reactivity is controlled by the use of Gd_2O_3 as burnable poison in specific fuel rods and movable absorbing elements belonging to the Adjust and Control System. Chemical compounds are not used for reactivity control during normal operation. The fuel cycle can be tailored to customer requirements, with a reference design for the prototype of 330 full-power days and 50% of core replacement.

Each Absorbing Element (AE) consists of a cluster of rods linked by a structural element (namely "spider"), so the whole cluster moves as a single unit. Absorber rods fit into the guide tubes. The absorbent material is the commonly used Ag-In-Cd alloy. Absorbing elements (AE) are used for reactivity control during normal operation (Adjust and Control System), and to produce a sudden interruption of the nuclear chain reaction when required (Fast Shutdown System).

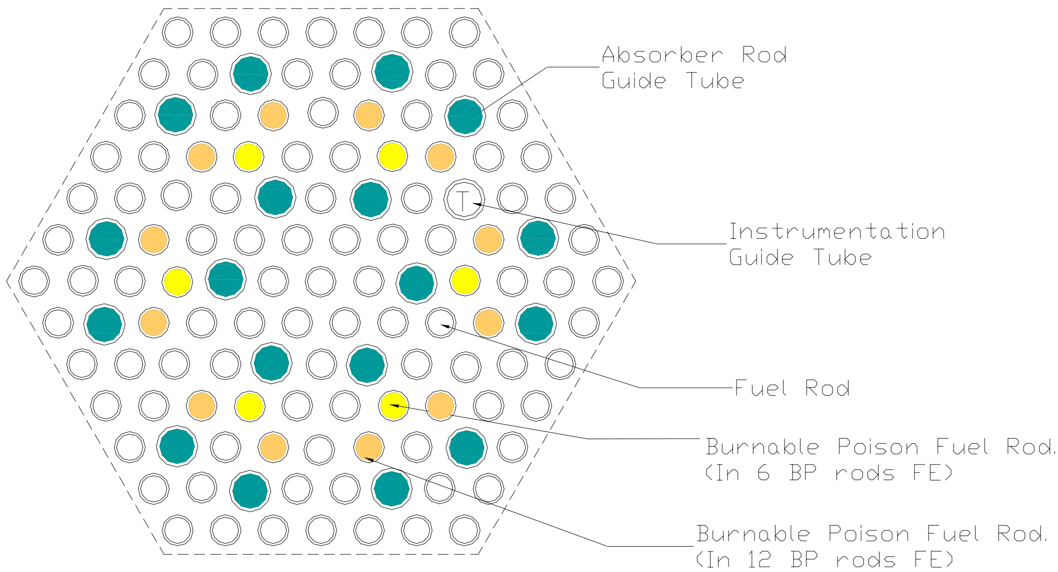


FIG. 6.2-2. Fuel assembly diagram. Fuel rods, guide thimbles and instrumentation thimble distribution.

6.2.2.3 Fuel handling and transfer systems

All the refueling tasks will be developed in the reactor hall, where storage space is allotted for the various components to be removed, while the RPV internals will be located in the Auxiliary Pool (Fig. 6.2-3).

Once the central drywell head, thermal insulation, pipe and instrumentation connections are removed, a seal is placed surrounding the RPV to allow the transfer channel flooding and to continue with the removal of RPV head cover and all the internals.

Finally, using an Operation Bridge, part of the fuel assemblies will be removed from the external zone of the core and placed in the Fuel Assemblies Pool, while the other part will be rearranged in the external zone and the new fuel assemblies will be placed in the central zone.

6.2.2.4 Primary components

Twelve identical 'Mini-helical' vertical steam generators, of the "once-through" type are placed equally distant from each other along the inner surface of the Reactor Pressure Vessel (RPV) (Fig. 6.2-4). They are used to transfer heat from the primary to the secondary circuit, producing superheated dry steam at 47 bar.

The secondary system circulates upwards within the tubes, while the primary goes in counter-current flow. An external shell surrounding the outer coil layer and adequate seal form the flow separation system. It guarantees that the entire stream of the primary system flows through the steam generators.

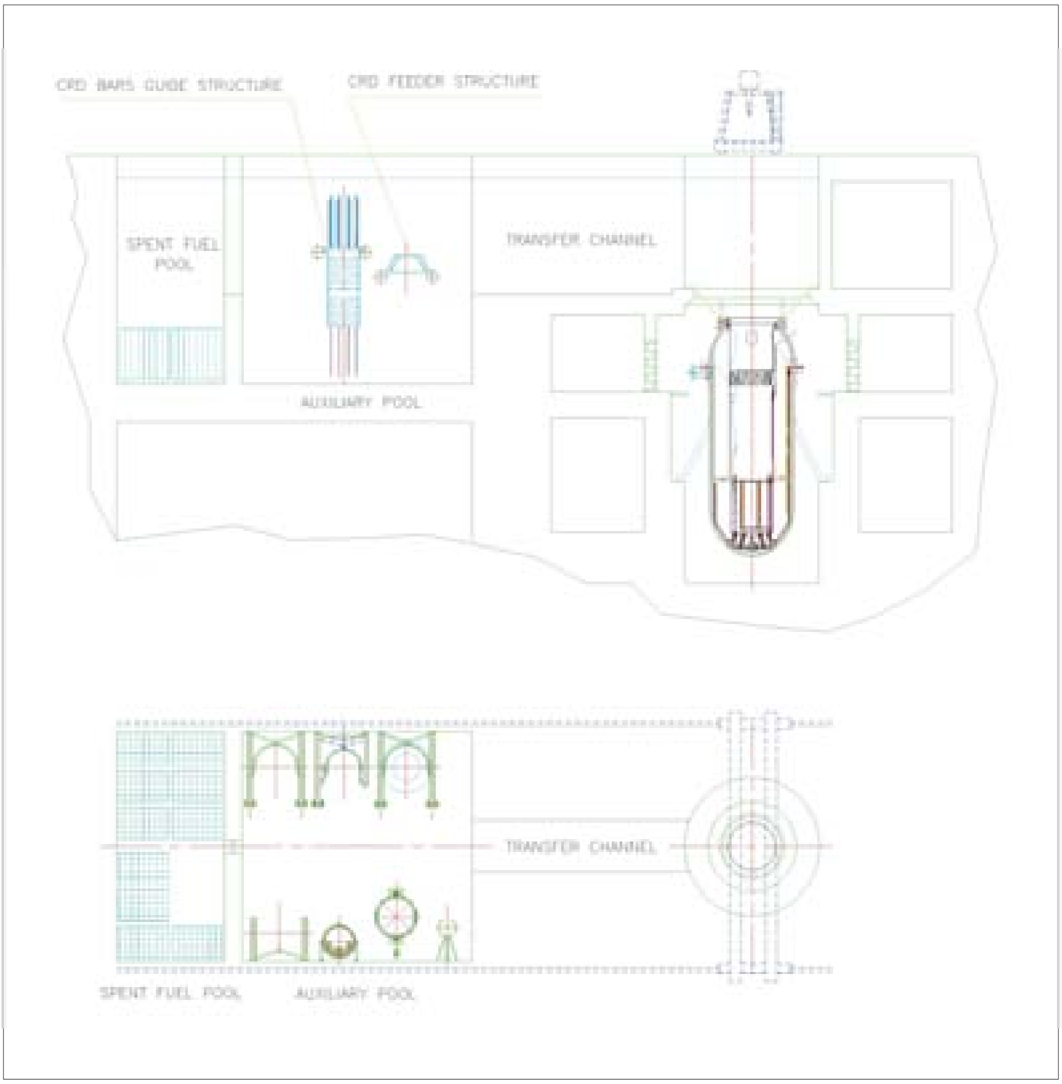


FIG. 6.2-3. RPV disassembly during refueling.

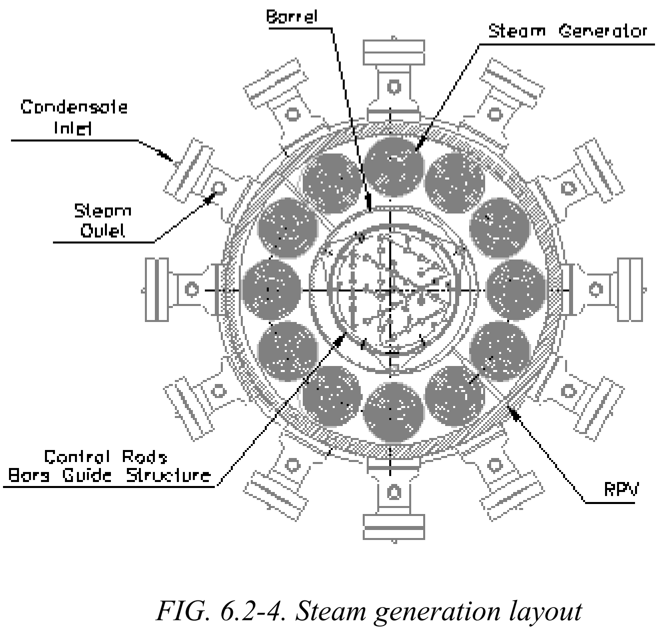


FIG. 6.2-4. Steam generation layout

In order to achieve a rather uniform pressure-loss and superheating on the secondary side, the length of all tubes is equalized.

Due to safety reasons, steam generators are designed to withstand the primary pressure without pressure in the secondary side and the whole live steam system is designed to withstand primary pressure up to isolation valves (including the steam outlet / water inlet headers) in case of SG tube breakage.

6.2.2.5 Reactor auxiliary systems

Figure 6.2-5 shows a diagram of the main reactor auxiliary systems.*Chemical / volume control system*
This system maintains a high degree of water purity within the RPV and allows controlling the water level while volume changes are produced by the operating conditions. The water removed from the RPV is cooled in a letdown heat-exchanger, undergoes a stage of pressure reduction, is treated in filters, resin beds, de-gasifier columns and returned to the RPV by the charging pumps through a regenerative heat exchanger. A control volume tank provides a volume reservoir that allows to contain all the water surplus of the RPV from the operation in solid way at 45°C until full power operation.

Suppression pool cooling and purification system

This system cools and purifies the suppression pool and the cooling pool for the residual heat removal system. The cooling system has redundancies: each branch has a heat exchanger and a pump, while both share the ion exchange bed for water purification.

In the event of a LOCA, this system is capable of feeding pure water into the RPV.

Shutdown cooling system

This system has two functions:
- It cools RPV water, removing decay heat during standard shutdown and refueling;
- It heats RPV water during plant start-up by an auxiliary steam system.

It is also redundant, each branch comprising a pump, plate heat exchanger for cooling, and shell and tube heat exchanger for heating.

Components cooling system - closed external circuit

The components cooling system supplies cooling water to the systems that may contain radioactivity, providing a barrier among the radioactive fluid and the closed external circuit. It is redundant and comprises pumps and heat exchangers.
The closed external circuit is also redundant. It has cooling towers and pumps.

Fuel pool cooling and purification system

It removes the heat resulting from nuclear decay of stored fuel elements and purifies pool water. The cooling system comprises two circuits - one in stand-by - each with a heat exchanger and a pump, sharing the filter and the ion exchange bed for water purification.

Control rod drive – hydraulic system

This system circulates water from the RPV to operate and maintain the Control Rods in position. It has two pumps in operation, to enhance system availability, as well as filters, valves for step-wise Control Rod motion and for operating Safety Rods rising, and redundant SCRAM valves.

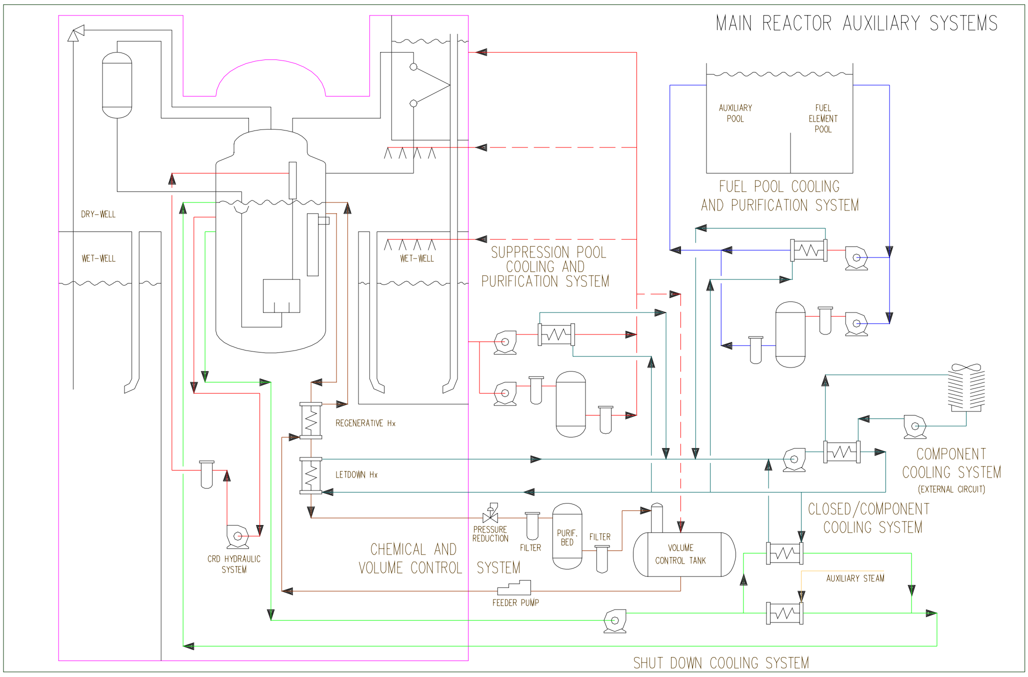


FIG. 6.2-5. Auxiliary systems.

6.2.2.6 Operating characteristics

For low power modules, the natural circulation of coolant produces different flow rates in the primary system according to the power generated (and removed). Under different power transients a self-correcting response in the flow rate is obtained [3].

Due to the self-pressurizing of the RPV (steam dome) the system keeps the pressure very close to the saturation pressure. At all the operating conditions this has proved to be sufficient to guarantee a remarkable stability of the RPV pressure response. The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in case of power ramps. The negative reactivity feedback coefficients and the large water inventory of the primary circuit combined with the self-pressurization features make this behaviour possible with minimum control rod motion. It concludes that the reactor has an excellent behaviour under operational transients.

6.2.3 Description of the turbine generator plant systems

The CAREM Plant has a standard steam cycle of simple design.

The steam generators are built as a drum-less “once-through” boiler without accumulators between them and the consumer equipment. In accordance with the behaviour of once-through boilers, steam is superheated in every plant condition. No super-heater is needed. Likewise, no blow-down is needed in the steam generators, reducing the waste generation.

The twelve steam generators are connected alternately in two groups of six to an annular collector. Each branch has its own relief and isolation valves and finally they are joined to deliver the steam to the turbine.

CAREM secondary circuit is not a safety-graded system; the nuclear safety of the plant does not rely on the functioning of the steam circuit.

6.2.3.1 Turbine generator plant

In CAREM commercial plants a two stages turbine with re-heater is used, and the exhaust steam at low pressure is condensed in a water-cooled surface condenser. In CAREM prototype a single turbine is used.

6.2.3.2 Condensate and feedwater systems

The condensate is pumped and sent to the full stream polishing system in order to maintain ultra-pure water conditions.

High purity water leaving the polishing system is sent to the low-pressure pre-heater using turbine extraction as a heating media. The warm water is delivered to the water accumulator in order to perform degassing operations with additional heating using extraction steam.

Water is then pumped to the high-pressure pre-heaters (two in tandem using extraction steam) and sent to the steam generators as a feed-water closing the circuit.

6.2.4 Instrumentation and control systems

6.2.4.1 Design concept, including control room

Control and supervision system

The control and supervision system is a "real time" computerized system for the control and supervision of the plant operation.

This system includes the control centers, the information processing centers, the man-machine interfaces, the automatic systems for process control, sensors, actuators and a communication net that connect these systems.

The main advantages obtained with the control and supervision system are:

- Distributed design,
- Functional separation of the process systems,
- Parallel task execution,
- Reliability improvement,
- Wiring cost reduction,
- Modular and flexible design,
- Adaptability to changes,
- Easy maintenance,
- Easy regulation,
- Possible expansion,
- Use of standard components,
- Simplification and parts saving,
- Lower reparation time.

The general architecture of the system has four hierarchic levels clearly identified for the processes and three communication levels.

The process levels are (Fig. 6.2-6):

- Supervision level: composed by a net of supervision nodes. At this level occur all the man-machine interactions between the operators and the supervision system;
- Information level: composed by a net of information nodes;
- Control level: Composed by a net of control units. These units constitute the connection with the field units;
- Field level: Composed by a net of field units that are the connection with the sensors and actuators and by all the sensors and actuators of the control and supervision system.

The three communication levels are at supervision net level, information net level and control net level.

The communication levels have been designed for in order to facilitate the data transmission between the net nodes spatially distributed. The main functions of the communication system are stratified according ISO standard.

Plant control

The operators have simple and direct access for commanding the control system.

The modules included in the control system of the plant can manage the regulation control circuits, the logical sequence execution and the manual control commands.

This system allows the operator, using a safety keyword, to set, modify and adjust the circuit parameters.

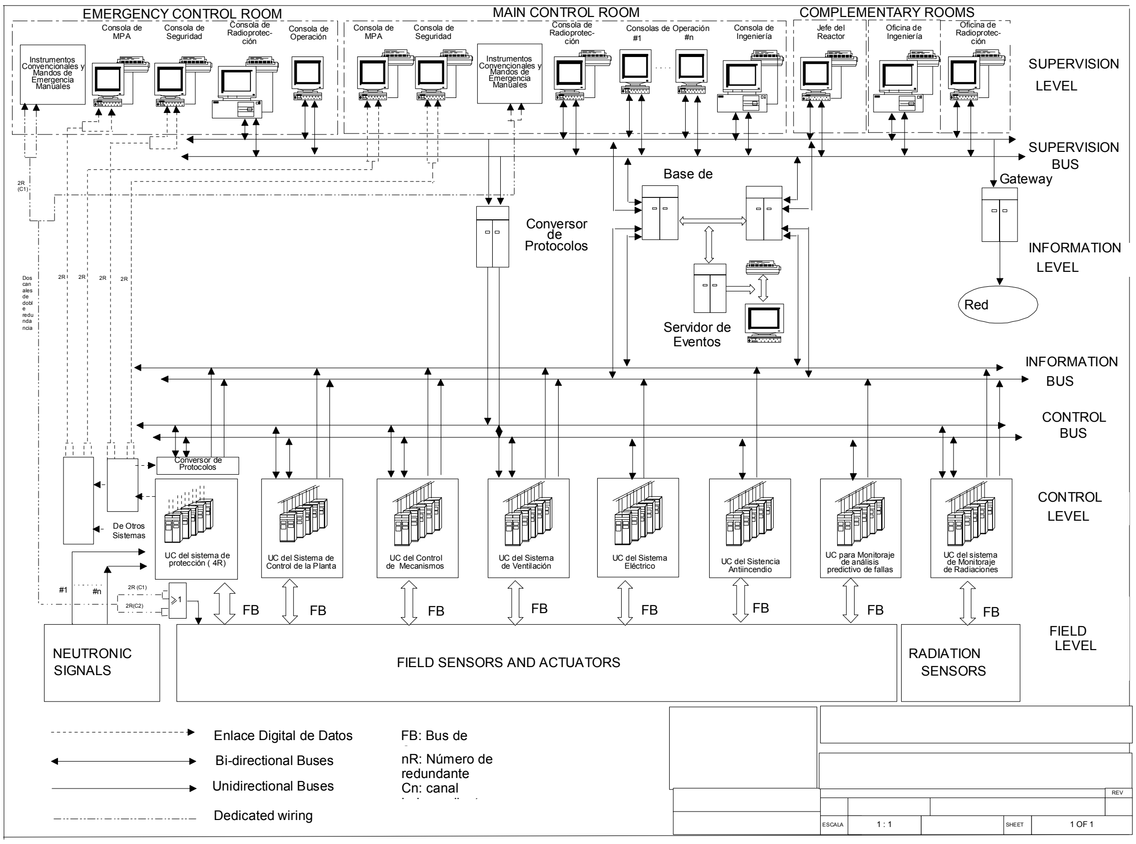


FIG. 6.2-6. Control system.

6.2.4.2 Reactor protection and other safety systems

The design of the reactor protection system was performed according with the most advanced technology for nuclear power plants design, the “defense in depth” principle and the early failure detection, with the object of avoid the occurrence of accidents beyond the design base.

The reactor protection system has two independent subsystems. The first subsystem, responsible of the generation of the first shutdown system trip signal, consists in a combination of hard logic and digital processing modules. The second subsystem, responsible of the generation of the second shutdown system trip signal, is based in a hard logic technology in order to fulfill with the diversity principle for the first and second shutdown systems.

The reactor protection system has four independent and redundant channels with voting and protective logic of dynamic type. This allows a high availability and reliability.

The main applied design criteria are:

- Physical and electrical independence,
- Functional diversity,
- Reduced size and robustness,
- Failure tolerance,
- Possible in operation testing,
- Safe failure.

The interaction between the protection system and the control system is performed through electrical isolation. The interfaces are design in order to avoid that any protection action could be inhibited by a control system action.

The design guarantees that once a protective action is initiated it will be completed.

6.2.5 Electrical systems

The electrical loads are divided in three classes:

- Class I: DC, no supply interruption is admitted
- Class III: AC, supply interruption is admitted during a certain period
- Class IV: AC, Supply interruption is admitted

Classes I and III correspond to the Safety-related system.

6.2.5.1 Operational power supply systems

Class IV includes all the conventional systems.

6.2.5.2 Safety-related systems

The electrical power supply corresponding to class I and III systems are distributed by two systems of independent buses.

This redounded system is separately connected to each bus with independent layout and connections. Both bus systems can be interconnected in case of failure.

Auxiliary generators will supply power to the essential systems in case of no power generation or external supply. These auxiliary generators are redounded, physically separated and they can supply each of the power distribution systems of classes I and III.

Class I is sized to supply power to selected safety-related loads for at least 48 hours before needing a connection to classes III, IV or other external power source.

6.2.6 Safety concept

6.2.6.1 Safety requirements and design philosophy

Emphasis has been given since the design genesis to prevention of core degradation accidents by means of passive safety features, guarantying no need of active systems or operator actions for a period of several days. A proper balance by design is assured to avoid jeopardising reactor economic competitiveness.

An innovative methodology to perform or assist reactor design, balancing safety and economics at the conceptual engineering stage is used in CAREM project. The key to this integral methodology is to take into account safety aspects in an optimisation design process where the design variables are balanced in order to obtain a better *figure of merit* related with reactor economic performance. The design parameter effect on characteristic or critical safety variables, chosen from reactor behaviour during accidents (*safety performance indicators*), is synthesised in *Design Maps*. These maps allow one to compare the observable with limits, which are determined by design criteria or regulations, and to transfer these restrictions to the design variables or parameters. Therefore, reactor dynamic response during transients or accidents and safety aspects are integrated by means of additional rules, to those necessary for steady state dimensioning, to the neutronic, thermal-hydraulic and mechanical areas in the conceptual engineering stage of the design. Therefore, by means of this methodology a simplified design can be obtained, compared to the resultant complexity when these concepts are introduced in a later engineering stage or as a "patch".

This methodology allows to balance and optimise the reactor itself and safety system in an early engineering stage in order to internalise cost-efficiently safety issues, based on the defence in depth approach, considering appropriate conservative assumptions and safety margins. Therefore, a balance between reactor inherent capability and safety systems to cope with the postulated initiating events is achieved.

Finally this balanced design prevents that the search for economic performance should cause less safe reactors and, likewise, guarantees the design competitiveness in spite of the unavoidable safety costs.

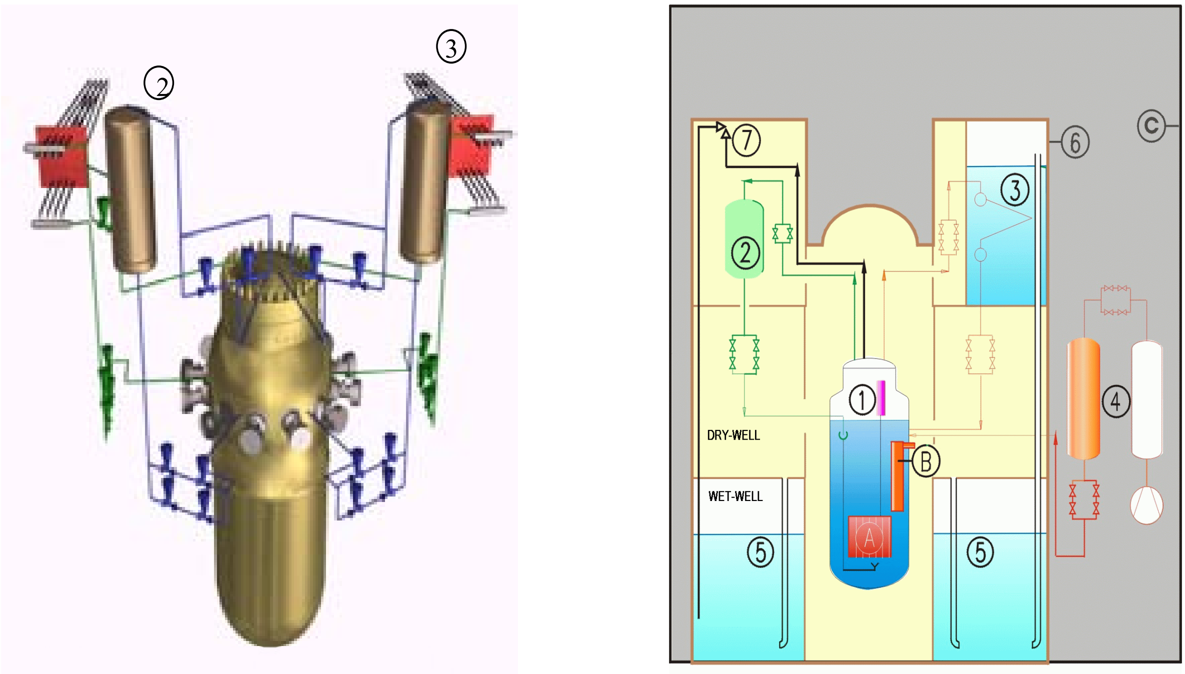
6.2.6.2 Safety systems and features (active, passive and inherent)

CAREM safety systems are based on passive features and must guarantee no need of active actions to mitigate the accidents during a long period (Fig. 6.2-7). They are duplicated to fulfil the redundancy criteria. The shutdown system should be diversified to fulfil Argentine regulatory body requirements.

The First Shutdown System (FSS) is designed to shut down the core when an abnormality or a deviation from normal situations occurs, and to maintain the core sub-critical during all shutdown states. This function is achieved by dropping neutron-absorbing elements into the core by the action of gravity. Each neutron absorbing element is a cluster composed of a maximum of 18 individual rods which are together in a single unit. Each unit fits well into guide tubes of each fuel assembly.

Hydraulic Control Rods Drives (CRD) avoid the use of mechanical shafts passing through RPV, or the extension of the primary pressure boundary, and thus eliminates any possibilities of large Loss of Coolant Accidents (LOCA) since the whole device is located inside the RPV. Their design is an important development in the CAREM concept [4]. Six out of twenty-five CRD (simplified operating diagrams are shown in Fig 6.2-8) are the Fast Shutdown System. During normal operation they are kept in the upper position, where the piston partially closes the outlet orifice and reduces the water flow to a leakage into the RPV dome. The CRD of the Adjust and Control System is a hinged device, controlled in steps fixed in position by pulses over a base flow, designed to guarantee that each pulse will produce only one step.

Both types of devices perform the SCRAM function by the same principle: "rod drops by gravity when flow is interrupted", so malfunction of any powered part of the hydraulic circuit (i.e. valve or pump failures) will cause the immediate shutdown of the reactor. CRD of the Fast Shutdown System is designed using a large gap between piston and cylinder in order to obtain a minimum dropping time thus taking few seconds to insert absorbing rods completely inside the core. For the Adjust and Control System CRD manufacturing and assembling allowances are stricter and clearances are narrower, but there is no stringent requirement on dropping time.



1: First Shutdown System
2: Second Shutdown System
3: Residual Heat Removal System
4: Emergency Injection System
5: Pressure suppression pool
6: Containment
7: Safety valves
A: Core
B: Steam Generators
C: Reactor Building

FIG. 6.2-7. Containment and safety systems.

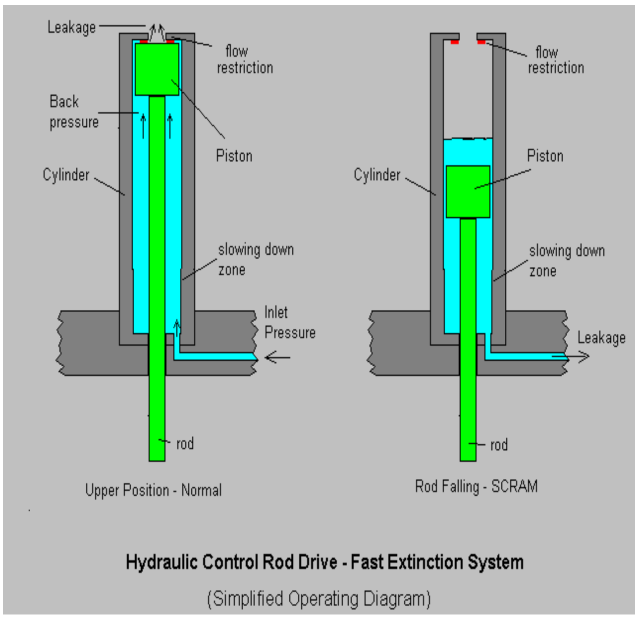


Figure 6.2-8 Simplified operating diagram of a hydraulic control rod drive (Fast Shutdown System)

The Second Shutdown System (SSS) is a gravity-driven injection device of borated water at high pressure. It actuates automatically when the Reactor Protection System detects the failure of the First Shutdown System or in case of LOCA. The system consists of two tanks located in the upper part of the containment. Each of them is connected to the reactor vessel by two piping lines: one from the steam dome to the upper part of the tank, and the other from a position below the reactor water level to the lower part of the tank. When the system is triggered, the valves open automatically and the borated water drains into the primary system by gravity. The discharge of a single tank produces the complete shutdown of the reactor.

The *Residual Heat Removal System (RHRS)* has been designed to reduce the pressure on the primary system and to remove the decay heat in case of Loss of Heat Sink. It is a simple and reliable system that operates condensing steam from the primary system in emergency condensers. The emergency condensers are heat exchangers consisting of an arrangement of parallel horizontal U tubes between two common headers. The top header is connected to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. The condensers are located in a pool filled with cold water inside the containment building. The inlet valves in the steam line are always open, while the outlet valves are normally closed, therefore the tube bundles are filled with condensate. When the system is triggered, the outlet valves open automatically. The water drains from the tubes and steam from the primary system enters the tube bundles and condenses on the cold surface of the tubes. The condensate is returned to the reactor vessel forming a natural circulation circuit. In this way, heat is removed from the reactor coolant. During the condensation process the heat is transferred to the water of the pool by a boiling process. This evaporated water is then condensed in the suppression pool of the containment.

The *Emergency Injection System* prevents core exposure in case of LOCA. The system consists of two redundant accumulators with borate water connected to the RPV. The tanks are pressurised, thus when during a LOCA the pressure in the reactor vessel reaches a relative low pressure, rupture disks break and the flooding of the RPV starts, preventing core un-covery for a long period. The R*esidual Heat Removal System* is also triggered to help to depressurise the primary system, in case the breakage area is small.

Three *safety relief valves* protect the integrity of the reactor pressure vessel against overpressure, in case of strong unbalances between the core power and the power removed from the RPV. Each valve is capable of producing 100% of the necessary relief. The blow-down pipes from the safety valves are routed to the suppression pool.

The primary system, the reactor coolant pressure boundary, safety systems and high-pressure components of the reactor auxiliary systems are enclosed in the primary containment - a cylindrical concrete structure with an embedded steel liner. The primary containment is of pressure-suppression type with two major compartments: a drywell and wetwell. The drywell includes the volume that surrounds the reactor pressure vessel and the second shutdown system rooms. A partition floor and cylindrical wall separate the drywell from the wetwell. The lower part of wetwell volume is filled with water that works as the condensation pool, and the upper part is a gas compression chamber.

A summary of functions to cover and the available safety systems is shown in Table 6.2-I.

For CAREM-25 accident analysis several initiating events were considered [5]:

Reactivity insertion accident: as the innovative hydraulic control drive for the Fast Shutdown System and the Adjust and Control System is located inside the RPV Rod Ejection Accident is avoided, only inadvertent control rod withdraw transients are postulated. Two scenarios considering FSS success and FSS failure with SSS actuation were modelled assuming conservative hypothesis. Simulation results show that safety margins are well above critical values (DNBR and Critical Power Ratio), no core damage is expected. Moreover, as there is no boron in the coolant, boron dilution as reactivity initiating event is precluded.

TABLE 6.2-I. SAFETY FUNCTIONS AND SAFETY SYSTEMS

Safety Function	Safety System
Reactivity Control	First Shutdown System: Safety control rods
	Second Shutdown System: Boron Injection
Primary Pressure Limitation	Safety Relief valves
	Residual Heat Removal System
Primary Depressurisation	Residual Heat Removal System
Primary Water Injection	High Pressure: Second Shutdown System
	Low pressure: Emergency Injection System
Secondary Pressure Limitation	Relief valves
Residual Heat Removal	Residual Heat Removal System

Loss of heat sink: in case of a total loss of feedwater to the steams generators, the Residual Heat Removal System is demanded cooling the primary system reducing reactor pressure to values lower than the ones of hot shutdown. In case of hypothetical failure of FSS, the reactor power reduces due to the negative reactivity coefficients without compromising the fuel elements. The SSS will guarantee medium and long-term reactor shutdown.

Total loss of flow: In natural circulation modules (Power lower than 150 MW(e)) there are no primary pumps, therefore this initiating event is excluded. In high power modules with forced circulation, natural circulation is enhanced intrinsically by the integral type reactor layout.

Loss of coolant accident: RPV penetration maximum diameter is limited by design, therefore no large LOCA is possible and there is no need of a high-pressure injection system. In case of LOCA the FSS, SSS, RHRS are demanded and when pressure decreases the *Emergency Injection System* discharge water to keep the core covered for several days. As no credit is given by design to active systems, the secondary system is not considered to cool and depressurise the primary system in safety evaluations, of course if it is available and in case of need it could be used as part of Accident Management Strategy. Moreover, by design no credit is given to a broken pipe as an injection line (steam coming into the RPV from the containment in case of high depressurisation of the primary system due to the use of the steam generators). The reactor inherent response to LOCA was also analyzed, considering FSS success and failure of all the Safety Systems related with core cooling. Due to the large water inventory over the core and the small penetration diameters through the RPV, the core uncovers after several hours.

Steam generator tube rupture: this accident is mitigated by isolating the group of steam generators affected, closing both the steam and feedwater lines. The secondary side of the steam generators reaches thermal equilibrium with the primary circuit, equalising pressure with this system. Eventually the reactor could continue operating at 50% of power.

Steam line break accident: The sudden depressurisation of the secondary side of the steam generators increase heat removal from the primary system with the consequent core overpower. Reactor shutdown (FSS and SSS) and Residual Heat Removal System are demanded and the reactor reaches a safe condition. In case of hypothetical failure of both shutdown systems, reactor overpower does no compromise safety critical values (DNB and CPR) because primary total heat removal by the steam generators is intrinsically limited by the reduced tube side water inventory.

Blackout: It is one of the events with major contribution to core meltdown probability in a conventional light water reactor. The extinction and cooling of the core and the decay heat removal are guaranteed without electricity by the passivity of safety systems. Loss of electrical power produces the interruption of the feed-water to the hydraulically driven CRDs, and thus produces the insertion of the absorbing elements into the core. Nevertheless in case of failure of the First and Second Shutdown Systems (both passive), in CAREM, feedback coefficients will produce the self-shutdown of the nuclear reaction without compromising safety related variables. The decay heat is removed by the Residual Heat Removal System with autonomy of several days.

As a general conclusion, it could be said that, due to the large coolant inventory in the primary circuit, the system has large thermal inertia and long response time in case of transients or accidents.

6.2.6.3 Severe accidents (beyond design basis accidents)

The CAREM concept highly enhances accident prevention and mitigation by simplicity, reliability, redundancy and passivity. Nevertheless, in case of the extremely low probability of failure of the passive safety systems (both redundancies) or no recovery actions after the design period to be covered by the passive safety systems (grace period -several days), a severe accident could be postulated to occur. Several features are considered to protect the confinement and address hypothetical severe accidents, allowing also the optimum use of all process systems for the primary cooling system and containment recovery after the grace period.

Severe accident prevention and mitigation features

- The absence of large LOCA prevents an early and sudden containment pressurisation, and together with the impossibility of a high reactivity insertion (no rod ejection) the possibility of a fast core melt is limited.
- Complementary and simple measures and accident management after the design period to be covered by the passive safety systems.
- Prevention of high-pressure core melt situation is ensured by means of the Residual Heat Removal System, complemented by relief valves opening.
- The suppression pool cooling and purification system cools and refills –if necessary- the suppression pool and the cooling pool for the residual heat removal system and feeds spray in the dry and wet-well to depressurise the containment. In the event of a LOCA, this system is capable of feeding pure water into the RPV.
- Provisions for injection of water to the reactor cavity from the refuelling water storage tank to cool the RPV from the out side to enhance the core debris, taking advantage of the high relationship between RPV lower bottom head area to core mass, characteristic of integral type reactors.
- When core uncovering is supposed, only for analysis proposes, low fuel elements heat-up rates in the uncovered part are predicted when the geometry is already intact, therefore core-melt characteristic time is large which eventually prevents temperature excursion due the metal water reaction, which in turns limits hydrogen generation rate.
- Reduction of the hydrogen-concentration in the containment by catalytic recombiners and if necessary selectively located igniters.
- Sufficient floor space for molten debris cooling.
- Extra layer of concrete to avoid containment basement exposure directly to debris.
- The suppression pool type containment provides a good physical mechanism for fission products retention by water.

6.2.7 Plant layout

6.2.7.1 Buildings and structures, including plot plant

CAREM nuclear island is placed inside a pressure suppression containment system, which contains the energy and prevents fission product release in the event of accidents.

The building surrounding the containment is placed in a single reinforced concrete foundation mat. It supports all the structures with the same seismic classification, allowing the integration of the RPV, the safety & reactor auxiliary systems, the fuel elements pool and other related systems in one block.

The plant building is divided in three main areas:

- Nuclear Module
- Turbine Module
- Control Module

6.2.7.2 Reactor building

Nuclear Module

This building acts as a Secondary Containment. The Containment itself is a free standing, vertical, cylindrical reinforced concrete structure, with flat head and bottom, designed to support pressure and temperature conditions, and act as a barrier to prevent fission product release to the Secondary Containment in the event of an accident.

The Nuclear Module has another relevant structural component in the shape of a box surrounded by 5 levels. In the upper part of this box are the Fuel Elements Pool and the Auxiliary Pool, and in the lower part are the Liquid Effluent and Spent Resin Pools. In CAREM-25 these five levels are:

Level + 15.20: The Reactor Hall where tasks related with the refuelling will be performed.

Level +10.00: The Exhaust stage of the HVAC system (Heating, Ventilation /Air Conditioning), the shielded rooms for filters and resins beds of the several water purification systems.

Level +5.20: The Gaseous Waste treatment system, valves rooms for the filters and resins beds, pumps and heat exchangers for the Components Cooling System.

Level 0.00: The connection between the Control and Turbine modules is here, as well as the emergency exit and the access to the Emergency Injection Systems. Workshops, compressors, and the HVAC injection equipment

Level -5.80: All the liquid effluents and spent resins collected are stored in shielded pools and treated at this level. Also the process equipment for the Reactor Auxiliary Systems like pumps and heat exchangers are housed in this area, with physical separation of equipment belonging to different redundancies.

6.2.7.3 Containment

The containment is divided in two main compartments: a drywell and a wetwell.

The upper drywell lodges the Second Shutdown System, the Relief valves and the headers of the Residual Heat Removal System. The emergency condenser pool is located at this level.

The central drywell houses the RPV and below it, separated by shielding, is the lower drywell.

The Peripheral Drywell surrounds the Central Drywell, below the upper level and houses the pipelines connected to the SG's.

The Wetwell (below the Peripheral Drywell, and surrounding the Central Drywell) is partially filled with water, conforming the Pressure Suppression Pool. The tube vents are immersed deep in the pool and they connect to the floor of the Peripheral Drywell.

6.2.7.4 Turbine building

Turbine Module

It houses the turbo-generator group, auxiliary services like de-mineralised water production system, chilled water, service steam, condensate polishing, and electric switchboards.

Close to the Reactor Module are the redundant diesel-generators with switchboards.

6.2.7.5 Other buildings

Control Module

The Control Complex is placed in this area of the building. It is formed by the Cables Room, Instrumentation Rooms, Main Control Room, the Remote Shutdown System (or Secondary Control Room), the Heating, Ventilating and Air Conditioning System for each Control Room and the offices for the operation personnel.

Rest rooms, lockers, Radiological Protection Facilities and the Safety Access System for personnel access lock to the Nuclear Module, are located in the basement.

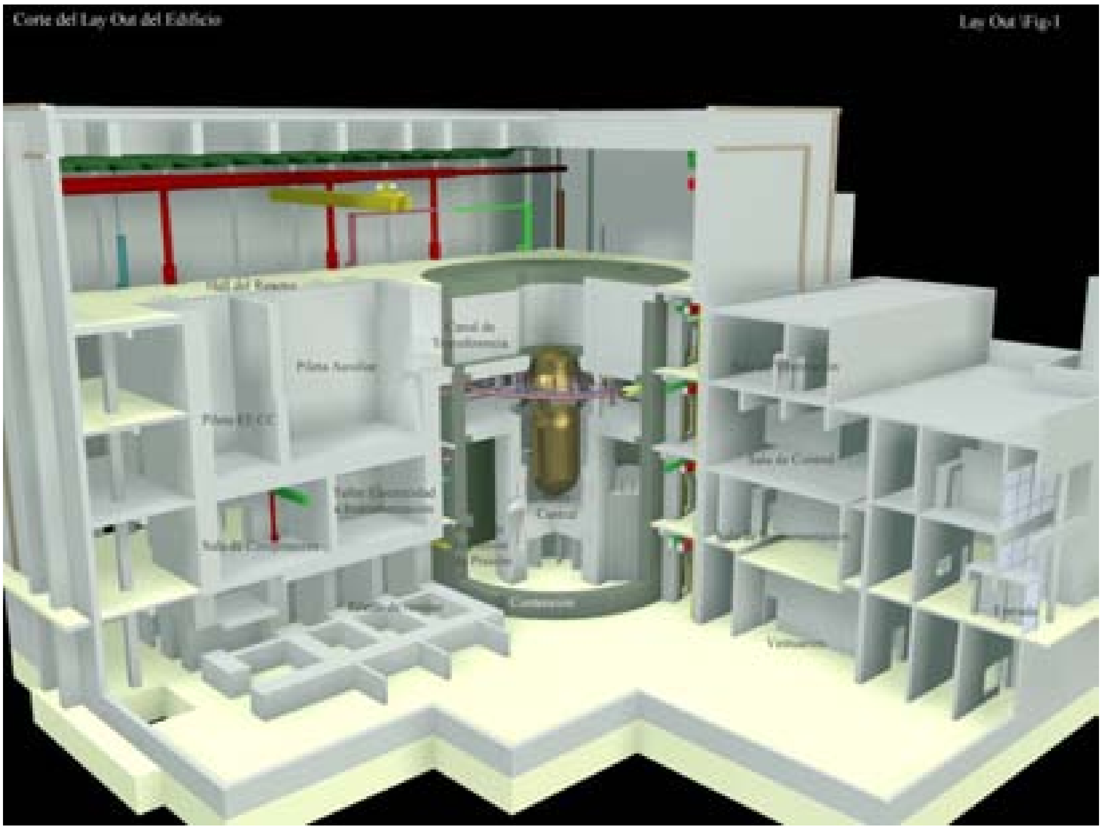


FIG. 6.2-9. Plant layout

6.2.8 Technical data

General Information

Design Name	CAREM 25	
Reactor Type	Integrated PWR	
Number of modules per plant	1	
Gross Thermal Power	100	MW(t)h
Electrical Power Output	27	MW(e)

Core and Reactivity Control

Fuel	Low Enriched UO_2	
Initial Enrichment	3.4	%
Average Power Density	55	kW/liter
Refuel Cycle	330 full power days	
Moderator	Light Water	
Clad Material	Zricaloy-4	
Number of Adjust and Control Rods	19	
Control rod assembly type	Cluster	
Control Rod neutron absorber	Ag-In-Cd	
Additional Shut-down system	Boron Injection	
Burnable poison	Gd_2O_3-UO_2	

Reactor Cooling System

Cooling Mode	Natural Circulation	
Coolant Inventory	39	m^3
Coolant mass flow through core	410	Kg/sec
Operating Coolant Pressure	12.25	MPa
Core Inlet Temperature	284	°C
Core Outlet Temperature	326	°C

Reactor Pressure Vessel

Overall Length	11	m
Inside Vessel Diameter	3.16	m
Average Vessel Thickness	135	mm.
Vessel Material	SA508 Grade 3 Class 1	
Lining Material	SS-304L	
Design pressure	14.5	MPa
Gross Weight (without internals)	130	Ton

Steam Generator

Number	12	
Type	Once through	
Configuration	Integrated – Vertical	
Tubes Material	Inconel 690 (SB 163 N06690)	
Shell Material	SS-304 L	
Thermal Capacity per SG	8.4	MW
Feed Water Temperature	200	°C
Steam Pressure	4.7	Mpa
Steam Temperature	290	°C

Containment

Type	Pressure Suppression	
Design pressure	0.55	MPa
Design Temperature	120	°C

First Shutdown System

Absorbing material	Ag-In-Cd

Assembly Type	Cluster	
Shutdown function driven by	Gravity	
Number of absorbing elements of the Fast Extinction System	6	
Number of absorbing elements of the Adjust and Control System	19	

Second shutdown system

Neutron Absorber Material	Borate solution	
Operation Mode	Gravity driven discharge	
Redundancy	Tanks 2 x 100 % Valves: 4 x 100 %	

Residual Heat Removal System-Emergency Condenser

Operation Mode	Steam Condensation	
Redundancy	Condenser 2 x 100 % Valves: 4 x 100 %	
Autonomy	> 48 hours	

Emergency Injection System

Pressure of Injection	1.5	Mpa
Operation Mode	Pressurized Tanks	
Redundancy	Tanks 2 x 100 % Valves: 4 x 100 %	
Autonomy	> 48 hours	

Safety Relief System

Pressure Set Point	14.0	Mpa.
Redundancy	3 x 100 %	

Turbine System

Type	Condensing	
Stages	1	
Speed	3000	rpm
Pressure	4.7	MPa
Temperature (30 °C superheated)	290	°C
Flow rate	175.32	Ton/hr

6.2.9 Measures to enhance economy and maintainability

Technical and economical advantages are obtained with the CAREM design as follows:

- In order to simplify the design the whole high-energy primary system, core, steam generators, primary coolant and steam dome, is contained inside a single pressure vessel. This considerably reduces the number of pressure vessels and simplifies the layout.
- Due to the absence of large diameter piping associated to the primary system, no large LOCA has to be handled by the safety systems. The elimination of large LOCA considerably reduce the needs in ECCS components, AC supply systems, etc.
- Due to self-pressurization, the elimination of an active pressuriser (heaters and sprinklers) results in lower costs and advantages for maintenance and availability.
- Eliminating primary pumps, in low power modules, results in lower costs, added safety, and advantages for maintenance and availability.
- The development of innovative hydraulic mechanism completely located inside the reactor pressure vessel eliminates the rod ejection accident. Furthermore, hydraulic control rod drive mechanism has a significantly lower cost compared with the a conventional PWR's control rod drive mechanism.
- Large coolant inventory in the primary system results in large thermal inertia and long response time in case of transients or accidents.
- The large water volume between the core and the RPV wall leads to a very low fast neutron dose to the wall.
- The design and fabrication approach is based on modularity. The reactor modules can be fabricated in a factory and be readily transported to the site, which reduces expensive on-site assembling/welding and ultimately construction time.
- Shielding requirements are reduced by the elimination of gamma sources of dispersed primary piping and parts.
- The ergonomic design and layout make the maintenance easier. Maintenance activities like the steam generator tubes inspection does not compete with refueling activities because it will be carried out from outside the vessel.
- The use of fewer active components increases plant availability and load factor.
- The reduction of staff and maintenance reduces the power generation cost.

6.2.10 Project status and planned schedule

The first step of this project is the construction of the prototype of about 27 MW(e) (CAREM-25). The conceptual engineering of this prototype reactor was completed.

An important experimental plan is underway for the First Shutdown System, or more specifically the Control Rod Drive Mechanism (CRD). The first series of design and verification tests have been conducted successfully in the Cold Low Pressure Rig (CEM) and second series of qualification tests is planned in the High Pressure Rig for CRD Test (CAPEM).

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6.3 SMART (KAERI, REP. OF KOREA)

6.3.1 Introduction

The SMART (System-Integrated Modular Advanced ReacTor) is an advanced integral Pressurized Water Reactor (PWR) that produces a rated thermal power of 330MW. The Korea Atomic Energy Research Institute (KAERI) completed the basic design of the SMART system in March, 2002. The concept of a nuclear desalination plant with the SMART is developed to supply 40,000 tons of fresh water per day and 90MW of electricity to an area with an approximate population of 100,000 or an industrialized complex. A detail design and construction project of a Pilot plant of the SMART, i.e., SMART-P project for demonstration of the relevant technologies, is currently underway and will be completed by 2008.

In the SMART design, all of the major primary system components, such as the fuel and core, twelve steam generators (SG), pressurizer and four main coolant pumps (MCP), are housed in a single reactor pressure vessel (RPV). This design feature excludes the possibility of the Large-Break Loss-of-Coolant-Accident (LB-LOCA). The SMART core design is characterized by an ultra long cycle operation with a single or modified single batch reload scheme, low core power density, soluble boron-free operation, enhanced safety with a large negative Moderator Temperature Coefficient (MTC) at any time during the fuel cycle, a large thermal margin, inherently free from xenon oscillation instability, and minimum rod motion for the load follow with coolant temperature control. The SMART MCP is a canned motor pump, which does not require pump seals. This characteristic basically eliminates small break Loss of Coolant Accidents associated with pump seal failure.

In addition to the above-mentioned inherent safety features and those coming from the innovative design features, further enhanced safety is accomplished with passive safety concepts. The engineered safety systems designed to function passively on the demand consist of a reactor shutdown system, passive residual heat removal system, emergency core cooling system, safety vessel, and containment overpressure protection system. Additional engineered safety systems include the reactor overpressure protection system and the severe accident mitigation system. The safety analyses for the SMART design have been performed and the results demonstrated that the key safety parameters of the limiting design base events do not violate the safety limits.

6.3.2 Description of the nuclear systems

6.3.2.1 Primary circuit and its main characteristics

The prominent design feature of SMART is the adoption of integral arrangement. All the primary components such as core, steam generators, main coolant pumps, and pressurizer are integrated into a single pressurized vessel without any pipe connection between those components. Figure 6.3-1 shows the structural configuration of the SMART reactor assembly. Four (4) main coolant pumps are installed vertically at the top of the reactor pressure vessel (RPV). The reactor coolant flows upward through the core and enters into the shell side of the steam generator (SG) from the top of the SG. The SGs are located in the circumferential space between the core support barrel and RPV above the core. The large volume at the top part of the RPV is used as a self-pressurizer. This integral arrangement of the major components into a single RPV is the most prominent difference in the design concepts of the RVA compared to the conventional loop type reactors.

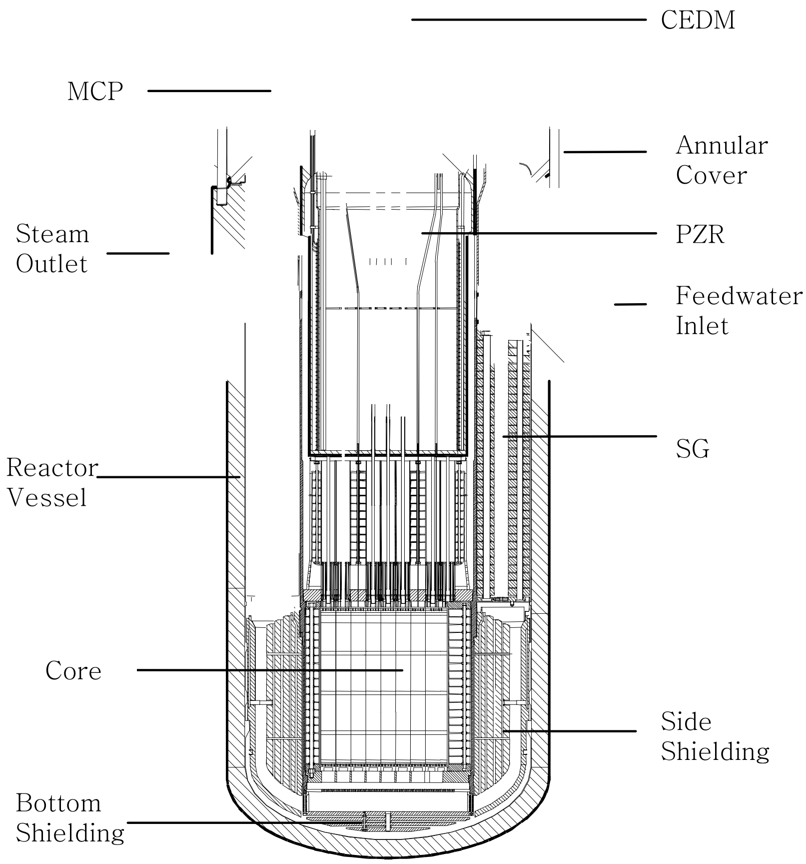

FIG. 6.3-1. The SMART reactor assembly

6.3.2.2 Reactor core and fuel design

The SMART core consists of fifty seven (57) fuel assemblies, which are based on the industry proven 17x17 square array of the Korea Optimized Fuel Assembly (KOFA). The SMART core design is characterized by an ultra long cycle operation with a single or modified single batch reload scheme, low core power density, soluble boron-free operation, enhanced safety with a large negative Moderator Temperature Coefficient (MTC) at any time during the fuel cycle, a large thermal margin, inherently free from xenon oscillation instability, and minimum rod motion for the load follow with coolant temperature control. The design characteristics with key design parameters of the SMART core are summarized in Table 6.3-I.

TABLE 6.3-I. DESIGN CHARACTERISTICS OF THE SMART CORE

Fuel type	17x17 UO_2 square FA	Excess reactivity (hot full power, eq. Xe, %Δρ)	2.96 (BOC), 2.40 (MOC) 1.41 (EOC)
Active fuel length (m)	2.0	Shutdown margin (cold zero power, %Δρ)	3.16 (BOC), 2.90 (MOC) 4.11(EOC)
Enrichment (w/o)	4.95	Maximum peaking factor (hot full power)	3.29
No. of fuel assemblies (FA)	57	Mod. temp. coeff. (hot full power, pcm/°C)	-72 < MTC < -42
Core power density (w/cc)		Fuel temp. coeff. (hot full power, pcm/°C)	-4.52 < FTC < -2.54
Refueling cycle (yr)	62.6	Scram worth (pcm)	29707(BOC) 28785(MOC) 31018(EOC)
No. of control element	> 3		
banks	49		
Control element material			
Burnable poison materials	Ag-In-Cd		
	Al_2O_3-B_4C, Gd_2O_3-UO_2		

6.3.2.3 Fuel handling and transfer systems

In the SMART refueling concept, a refueling machine, fuel location indexing machine, and shielding shell are involved. A fuel location indexing machine is mounted on the annular cover of the RPV. This device permits the fixing of a telescope grip to the individual fuel assembly. A shielding shell is also installed to protect the personnel and reactor vessel upper guide structure from the radiation contamination during the refueling process. The spent fuel is withdrawn from the core and held by the refueling machine. A crane then carries the machine to the spent fuel area where spent fuel assemblies will be stored. Fresh fuel assemblies are loaded in compliance with the core loading program. Fuel assemblies are placed in the designated core cells by means of the same fuel location indexing machine.

6.3.2.4 Primary components

Reactor Pressure Vessel (RPV)

The SMART RPV is a pressurized cylindrical vessel accommodating all major components of the primary system. The RPV consists of a cylindrical shell with an elliptical bottom and an upper flange part welded to the shell. The RPV cover consists of two parts: the annular peripheral part and the round central one which serves as the cover of the in-vessel pressurizer.

The annular cover is fixed on the vessel flange by means of a stud bolt joint. The vessel-to-annular cover joint is made leaktight by a welded torus sealing. The central cover is fastened to the annular one by a flangeless joint, using a rack-and-gear mechanism.

All penetrations are limited in the vessel head region except for SG feedwater and steam nozzles to assure an ample amount of coolant inventory to eliminate core uncovery in any case of postulated pipe break accident. On the annular cover, there are four seats for MCPs, make-up piping nozzles, resistance thermometers, branch pipes, etc. On the outer surface of the central cover, there are nozzles for installing 49 CEDMs, rack and gear drives, branch pipes, etc.

Reactor Internals

The reactor internal structures are located in the RPV and divided into three parts: core support barrel (CSB) assembly, side screen assembly, and upper guide structure (UGS) assembly.

The CSB assembly provides a support structure for the core and a suitable flow path within the reactor pressure vessel to guide the primary coolant through the core. The core shroud assembly provides an envelope within the core support barrel to direct the coolant flow through the core, and minimize bypass flow around the core.

The side screen assembly composed of multiple cylindrical shell structures provides a shield for the reactor pressure vessel against radiation damage by the neutron flux. The assembly also supports the steam generator cassettes.

The UGS assembly guides the control elements into the core and maintains the core in lateral and azimuthal alignment in conjunction with the CSB assembly. The assembly also provides suitable flow guidance from the core to the CSB outlet nozzles and limits lifting the core during conditions which cause upward force on it.

Steam Generator (SG)

Twelve identical SG cassettes are located in the annulus formed by the RPV and the core support barrel. Each SG cassette is a once-through type with helically coiled tubes wound around the inner shell. The primary coolant flows downward in the shell side of the SG tubes, while the secondary feed-water flows upward in the tube side. Therefore, the tubes are under compressive loads from the greater primary pressure, reducing the stress corrosion cracking and thus reducing the probability of tube rupture. The steam exits the SG with 40°C superheated steam condition at normal operation and thus a steam separator is not required. Three (3) steam and feedwater pipes from the adjacent steam generator cassettes are connected together to form a section. There are thus a total of four (4) sections in SMART. If there is a leakage in one or more of the tubes, the relevant section is isolated and SMART can be operated with reduced power until the scheduled shutdown. With the adoption of a modular concept, any defective SG can be replaced individually.

Pressurizer (PZR)

The SMART PZR is an in-vessel self-controlled pressurizer located in the upper space of the reactor assembly and is filled with water, steam and nitrogen gas. The self-pressurizing design eliminates the active mechanisms such as spray and heater. The system pressure is determined by a sum of the steam and nitrogen partial pressures. To minimize the contribution of the steam partial pressure, a PZR cooler is installed to maintain the low PZR temperature, and wet thermal insulator is installed to reduce the heat transfer from the primary coolant. The variation of the core outlet temperature is programmed so that the system pressure variation becomes minimized by canceling out the increase of the coolant volume of the hot part with the decrease of the coolant volume of the cold part during power increase (maneuvering). Thus system pressure is maintained almost constant during power maneuvering.

Control Element Drive Mechanism (CEDM)

Due to soluble boron-free operation, an important design requirement for the SMART CEDM is a fine maneuvering capability to control the excess core reactivity. The conventional magnetic jack-type CEDM is not able to meet this requirement. Thus, linear step motor type CEDM with easy access for in-service inspection and replacement is employed. The minimum step length is 4mm per step that is short enough for the fine reactivity control. Forty-nine (49) CEDMs are installed in the fifty-seven(57) fuel assemblies of the SMART core. There may exist electro-magnetic interference between CEDMs due to crowded installation on the limited space of the RPV head, and thus accidental mal-function may be induced. However, the magnetic-field effect analysis shows that the maximum field density is about 4.7×10^{-6} Tesla that is far below of the required field density, 1.5×10^{-3} Tesla for the position indicator operation, and thus no electro-magnetic interference effect between CEDMs exists.

Main Coolant Pump (MCP)

The SMART MCP is a canned motor type pump that eliminates the problems of conventional seals and associated systems. In other words, canned motor type pump eliminates a small break loss of coolant accident (SBLOCA) associated with a pump seal failure. Four (4) MCPs are installed vertically on the RPV annular cover. MCP is an integral unit consisting of a canned asynchronous 3-phase motor and an axial flow single-stage pump. The motor and pump are connected through a common shaft rotating on three radial and one axial thrust bearings. The impeller draws the coolant from above and discharges downward directly to the SG. This design minimizes the pressure loss of the flow.

6.3.2.5 Reactor auxiliary systems

Component Cooling System (CCS)

The component cooling system removes the heat generated in the MCPs, CEDMs, PZR, and the internal shielding tank.

Purification System (PUS)

The purification system purifies the primary coolant and control water chemistry to provide reliable and safe operation of the reactor core and all equipment in any mode of power plant operation.

6.3.2.6 Operating characteristics

The SMART core is designed to operate at a maximum core power level of 339 MWt. The turbine bypass system and condenser, in conjunction with the power cutback system, can accommodate a 100% load rejection without a reactor trip and without lifting either primary or secondary safety valves.

The maneuvering capability of the SMART design includes the following;

- Step turbine power changes of ±20% in the 20% - 100% power range;
- Power ramp of 5% per minute in the 20% - 100% power range;
- 100% load rejection without a reactor trip.

6.3.3 Description of the turbine generator plant systems

6.3.3.1 Turbine generator plant

The reference concept of the turbine plant has been developed including a coupling system for the seawater desalination system. The overall design is similar to that of present-day power plant. The turbine plant receives superheated steam from the NSSS. It uses most of the steam for electricity generation, seawater desalination and pre-heaters. The SMART desalination system consists of 4 units of Multi-Effect Distillation (MED) combined with a thermal vapor compressor (MED-TVC). Each Distillation Unit is capable of producing 10,000 m^3/day of distilled water for 24 hours of operation at the maximum brine temperature of 65°C and the supplied sea water temperature of 33°C. The MED-TVC unit is designed with a performance ratio (PR) of 15 and a motive steam to load ratio of one. The PR and steam to load ratio were determined based on the results of the thermodynamic analysis and economic evaluation for the water production capacity of 40,000m^3/day and electricity generation of 90 MWe.

SMART and MED-TVC units are connected through the steam transformer. The steam transformer produces the motive steam using extracted steam from a turbine and supplies the process steam to the desalination plant. It also prevents the contamination of the produced water by hydrazine and radioactive material of the primary steam.

6.3.3.2 Condensate and feedwater systems

The condensate and feedwater systems are similar with those of the present-day plants consisting of condensate pumps, feedwater purification system, low-pressure and high-pressure preheaters, main feedwater pumps, startup feedwater pumps and feedwater control valve. In the SMART plant, where once-through steam generators are utilized, change of the feedwater flow rate leads the change of the steam flow rate. The main steam pressure is controlled so as to be constant during power operation. The plant load change is achieved by changing the feedwater flow rate.

6.3.3.3 Auxiliary systems

Auxiliary systems for the turbine generator plant system consist of condensate polishing system, service air systems, nitrogen system, drain systems, HVAC systems, chilled water systems, wastewater treatment system, component cooling water system, etc.

6.3.4 Instrumentation and control systems

6.3.4.1 Design concept, including control room

The SMART control systems are mainly composed of a primary control system, secondary control system and process control system. The primary control system performs mainly reactor regulating control and CEDM control. The secondary control system includes the main steam pressure control system, feedwater control, and turbine control system. The process control system manages various pumps and valves that are needed to maintain the normal operating condition of SMART.

The control systems are designed with a layered structure based on digital and information processing technology. Each layer is functionally and locally distributed. Also to improve maintainability, the control systems adopt module-based design and standardized equipment. To improve availability and reliability of the control systems, hot stand-by and duplex (redundancy) structures are introduced into each system.

Sensors and instrumentation system

The sensors and instrumentation systems perform the functions of acquisition of plant parameters, signal processing, and signal transmission to I&C systems such as the protection system, control system, and monitoring system. The high reliability and performance of sensor and instrumentation systems is achieved by the following advanced features:

- Use of digital technology in the instrument loops;
- Use of remote multiplexers at local field to achieve reduction of cabling;
- Use of digital signal processing techniques such as signal validation and fault diagnostics;
- Adaptation of a sharing sensing signal concept for protection & control system;
- Installation of in-vessel safety-grade neutron detectors & digitization of signal processing electronics.

Data Communication System (DCS)

The DCS provides highly reliable data exchanges between intra and inter systems. The DCS architecture is developed for SMART based on the communication network technology. The design philosophy for the safety-classified DCS is:

- Status-based transmission,
- Deterministic architecture,
- Full separation and isolation,
- Highly reliable and hard real-time features,
- Adaptation of verifiable technologies.

The SMART data communication is characterized by the following design features;

- Fixed access controls using time division multiplexing and switching methods;
- Fiber optic transmission medium;
- Separation & isolation between transmission and receiving path;
- Point-to-point link architecture;
- More than 10 Mbps for each link connection;
- Critical fault management.

Information Processing Design

The information level in the SMART control room consists of a large-scale display panel (LDP), alarm and indication system (AIS), information processing system (IPS), and soft-controller (SC). The purpose of the information level is to provide the plant operating staff with various supervisory monitoring aids every one second via LDP, FPDs, and CRTs and enable them to enhance the plant operability and lead the plant to safe states. Safety parameters, functions and alarms are presented in a hierarchical display manner and information is well organized with the optimized navigation concept.

The availability of the information level is more than 99.5% by adopting the primary and hot-standby redundant concept for data processing servers and distributed graphic processing clients through a standard network. The heartbeat interconnection between the primary and hot-standby processor is used to detect a failure so that bumpless operating is achieved.

Control room design

The SMART compact control room adopts advanced information display and soft-control technologies such as FPD (Flat Panel Display), CRT, and LSDP (Large Scale Display Panel). Their flexible capabilities of displaying information are promising factors which can improve operators' performance by providing them with graphical user interface for monitoring and controlling the plant system in a task compatible manner.

The SMART compact control room is designed for one man operation (staffing goal) under normal conditions of the plant. The SMART compact control room encompasses LSDP, MCB (Main Control Board), Safety Console, and Supervisor's Console. The following are their features to support the operating staff. LSDP provides operators with overall plant information for determining the current process and safety status of the plant. It serves operators by formulating the common mental model of the plant status. MCB, a seated-type compact workstation consisting of CRTs, FPDs, keyboards, and pointing devices, provides reactor/turbine operators with a means of monitoring and controlling the plant processes.

The safety console provides the operator with safe shutdown capabilities in the event of failures of the MCB. It also contains a set of hardwired switches for manual actuation of ESF equipment and reactor trip.

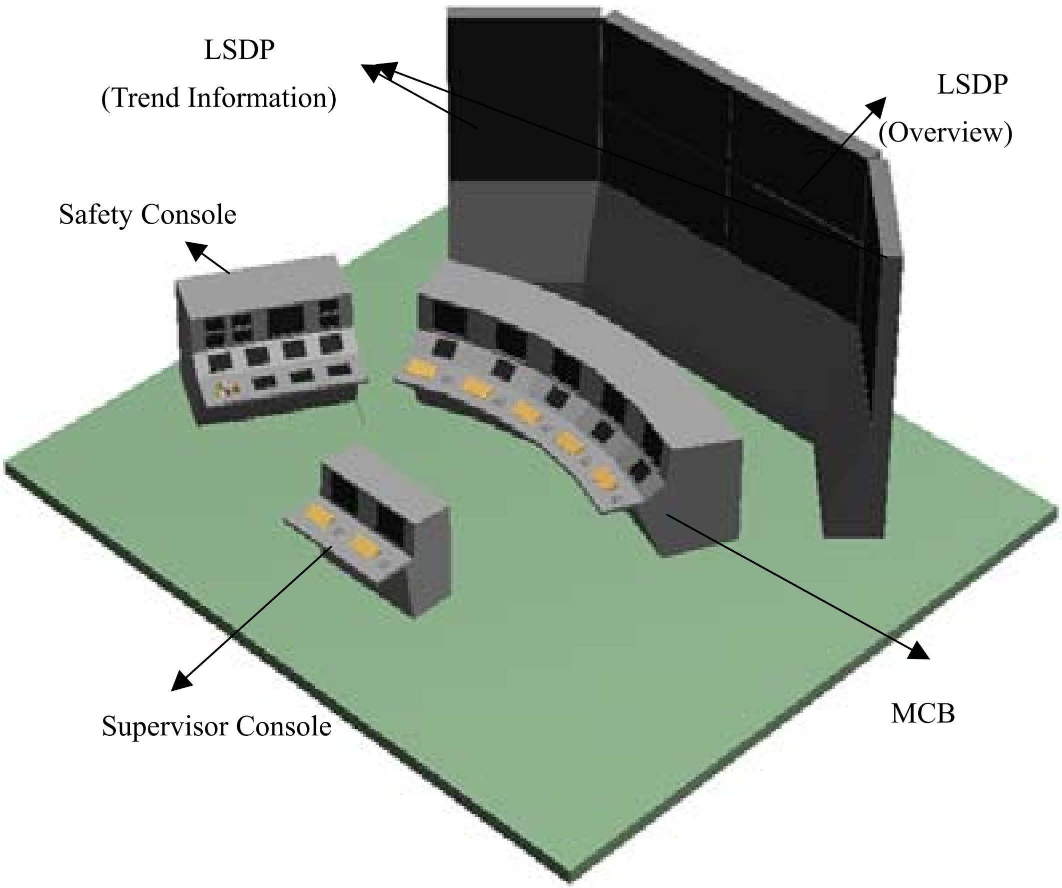

FIG. 6.3-2. Compact control room

6.3.4.2 Reactor protection and other safety systems

The protection system of SMART includes two major systems: the reactor protection system and the actuation system of engineered safety features. The basic structure of the system, which is responsible for performing the main functions, consists of four redundant trip channels, which are modularized by specific functions and segmented for functional diversity, respectively, and uses two out of four decision logic.

The safety-classified digital protection system automatically shuts the reactor down whenever conditions reach the safety setpoint, and actuates the engineered safety systems to mitigate the consequences of accidents. Control grade diverse means are additionally provided to prevent the Anticipated Transient Without Scram (ATWS).

The SMART protection system is characterized by the following design features:

- Adaptation of digital technologies and equipment (Digital Signal Processor (DSP) on VERSA Module Eurocard (VME) back-plane), which considerably improves the performance and flexibility;

- Use of a highly reliable fiber optic network for intra and inter communication;
- Use of a self-diagnosis function to assist the system performance;
- Use of a automatic on-line test function to increase system availability.

6.3.5 Electrical systems

The concept of the electrical systems is basically the same as those of the PWRs operating in Korea. The electrical systems of the SMART include the main generator, main transformer, unit auxiliary transformers, stand-by auxiliary transformers, diesel generators, and batteries. The electrical systems, including the Class 1E and the non-class 1E, are based on a “two train” approach.

6.3.5.1 Operational power supply systems

The main generator is connected to the grid via the main transformer. The unit auxiliary transformers are connected between the main generator and main transformer. The stand-by auxiliary transformers (the offsite power source) receive electrical power from the grid. The unit auxiliary transformers and/or the stand-by auxiliary transformers supply the normal electrical power source to the plant equipment for startup, normal operation and shutdown.

If the normal electrical power is not available, the diesel generators (non-Class 1E, Class 1E) can provide a back-up source of electrical power. In the event of a station blackout (loss of offsite and onsite AC power supply), the alternate AC diesel generator (class 1E) supplies AC power to the Class 1E loads to maintain the reactor in a safe shutdown condition. The vital batteries have the adequate capacity to provide the necessary DC power to perform the required functions in the event of an accident assuming a single failure.

6.3.5.2 Safety-related systems

The electric power for the safety related systems are supplied by the following sources:

- The normal power source using the electric power of the main generator;
- The off-site power source connected through the stand-by transformers;
- The on-site stand-by power source using the diesel generators;
- The power source of the alternative AC diesel generator.

To ensure the safety of the reactor, the electric power for the safety related systems are designed to provide a highly reliable source of power supply (Class 1E). Two physically separated power sources are provided to the safety related system.

6.3.6 Safety concept

6.3.6.1 Safety requirements and design philosophy

The SMART design combines firmly established commercial reactor design technologies with new advanced technologies. Thus substantial parts of the design features of the SMART have already been proven in industries such as the core using the Korea Optimized Fuel Assembly (KOFA). The new and advanced features of the SMART design provide significant enhancements in safety. The sources of safety enhancements can be classified into three major categories as follows;

- Innovative design features,
- Inherent safety design features,
- Passive engineered safety systems.

The SMART is an integral type PWR and one reactor pressure vessel contains major primary components, such as modular once through steam generators, canned motor main coolant pumps, self-pressurizing pressurizer, etc. This design feature excludes the possibility of the Large-Break Loss-of-Coolant-Accident (LB-LOCA) by the elimination of coolant loops. The integral reactor type feature also reduces the fast neutron fluence on the Reactor Pressure Vessel. Additional features include the canned motor pumps, which remove the necessity of pump seals and the possibility of the small-break LOCA associated with pump seal failure, the passive pressurizer which does not have active spray and heater. This pressurizer design feature eliminates the complicated control and maintenance requirements and reduces the possibility of malfunction. Compared with the defense-in-depth concept of conventional reactors, the SMART is reinforced by an additional physical boundary of the safety vessel for the protection of the public safety from radiation releases. Normally the defense-in-depth capability of conventional reactors includes multiple levels of defense like the fuel cladding boundary, the reactor pressure boundary, and the containment pressure boundary. Advanced man-machine interface systems using digital technologies and equipments will reduce the human error factors.

There are many inherent safety features in the SMART design. Those include a large negative moderator temperature coefficient due to the boron-free operation concept, a low core power density. The small core size eliminates the possibility of the xenon oscillation instability too.

Besides the inherent safety characteristics of SMART, further enhanced safety is accomplished with highly reliable engineered safety systems. The engineered safety systems designed to function passively on the demand consist of a reactor shutdown system, passive residual heat removal system, emergency core cooling system, safety vessel, reactor overpressure protection system and containment overpressure protection system. With all those enhanced safety features, the core meltdown frequency is predicted by PSA to be $8x10^{-7}$. Even after the beyond-design accidents accompanying core damage, the safety vessel of the SMART provides an additional barrier to the radioactive release compared with the current commercial reactors. Also the water inside the internal shielding tank surrounding the bottom-side of the RPV behaves as an external cooling mechanism to mitigate severe accidents.

6.3.6.2 Safety systems and features (active, passive and inherent)

Besides the inherent safety characteristics of SMART, further enhanced safety is accomplished with highly reliable engineered safety systems. The engineered safety systems designed to function passively on demand consist of a reactor shutdown system, passive residual heat removal system, emergency core cooling system, safety vessel, and containment overpressure protection system. Additional engineered safety systems include the reactor overpressure protection system and the severe accident mitigation system.

Reactor shutdown system

The shutdown of SMART can be achieved by a function of one of two independent systems. The primary shutdown system is the control rods containing Ag-In-Cd absorbing material. The shutdown signal de-energizes the CEDM and then the control rods drop into the reactor core by the force of gravity and immediately stop the neutron chain reactions.

In the case of failure of the primary shutdown system, the emergency boron injection system is provided as an active back-up by make-up pump. One train of make-up system which is designed as safety grade system, is sufficient to bring the reactor to the sub-critical condition.

Passive Residual Heat Removal System (PRHRS)

The system passively removes the core decay heat and sensible heat by natural circulation in case of an emergency such as unavailability of feedwater supply or station black out. Besides, the PRHRS may also be used in case of long-term cooling for repair or refueling. The PRHRS consists of 4 independent trains with 50% capacity each. Two trains are sufficient to remove the decay heat. Each train is composed of an emergency cooldown tank, a heat exchanger and a compensating tank.

The system is designed to keep the core un-damaged for 72 hours without any corrective actions by operators at the postulated design basis accidents. In the case of a normal shutdown of SMART, the residual heat is removed through the SG to the condenser with a turbine bypass system.

Emergency Core Cooling System (ECCS)

The SMART design excludes any possibility of large break Loss of Coolant Accident. The largest sized pipes connected to the outside of the RPV are20 mm. The ECCS is thus provided to protect the core uncovery by mitigating the consequences of design basis events such as small break LOCA through make-up of the primary coolant inventory.

When an initiating event occurs, the primary system is depressurized, the valve in the line of the ECCS is automatically opened and the water immediately comes into the core by gas pressure.

The ECCS consists of two independent trains with 100% capacity each. Each train includes a cylindrical water tank pressurized with nitrogen gas, isolation and check valves, and a pipe of 20 mm in diameter connected to the RPV. The system provides vessel refilling so that the residual heat removal system can function properly in the long-term recovery mode following the event.

Safety vessel

The safety vessel is a leak-tight pressure retaining steel-made vessel intended for the accommodation of all primary reactor systems including the reactor assembly, pressurizer gas cylinders, and associated valves and pipings.

The primary function of the safety vessel is to confine the radioactive products within the vessel and thus to protect any primary coolant leakage to the containment. The vessel also has a function of keeping the reactor core undamaged for 72 hours without any corrective actions at the postulated design basis accidents including LOCA, with the operation of the PRHRS and ECCS. The steam released from the opening of the relief valve of the safety vessel at the postulated beyond design basis accidents is sparged into the external shielding tanks and immediately condensed.

Containment Overpressure Protection System (COPS)

The containment is a steel structure with a concrete building enclosing the safety vessel to confine the release of radioactive products to the outside environment in postulated beyond design basis accidents relating to the loss of integrity of the safety vessel.

In any accident causing a temperature rise and thus a pressure rise in the containment, containment cooling is done, in a passive manner, by removing the heat from the containment. The heat is removed through the steel structure itself, and through the emergency cooldown tanks installed inside the containment.

A rupture disc and a filtering system are also provided in the containment to protect the steel structure from overpressure and to purify the released radioactive products at the postulated beyond design basis accidents.

Reactor Overpressure Protection System (ROPS)

The function of the ROPS is to reduce the reactor pressure at the design basis accidents related with a control system failure. The system consists of three pilot operated safety relief valves (POSRVs), which are installed on three gas cylinders respectively. The steam discharge lines of the POSRVs are combined to a single pipeline and connected to the external shielding tank.

When the primary system pressure increases over the setpoint, POSRVs are opened to discharge the steam into the external shielding tank through a sparging device.

Severe Accident Mitigation System (SAMS)

The function of the SAMS is to prevent the egress of molten corium resulting from a severe accident out of the containment.

The egress of corium can be avoided due to the design characteristics of the safety vessel and containment together with the operation of the safety systems. A small air gap under the RPV is filled with water from the makeup system at the severe accident. The in-vessel cooling prevents the egress of the corium out of the RPV. In addition, the water in the internal shielding tank provides the external cooling of the RPV and prevents the egress of the corium out of the RPV.

Hydrogen igniters are provided in the safety vessel to remove the explosive hydrogen generated during a severe accident.

6.3.6.3 Severe accidents (beyond design basis accidents)

No information provided.

6.3.7 Plant layout

6.3.7.1 Buildings and structures, including plot plan

Figure 6.3-3 shows the general arrangement of an SMART plant. The building arrangement is designed considering simplification and convenience of maintenance. Approximate sizes of the some major buildings are:

Building \ Type, Size	Type	L x W (m)	H (m)	Area (m^2)	Volume (m^3)
Containment Building	Steel/Concrete Double Containment	18 x 16	45	576	14,079
Turbine Building	Steel Structure	41 x 28	27.5	4,100	31,570
Auxiliary Building	Reinforced Concrete Structure	45 x 34	34	5,598	38,556
Utility Building	Steel Structure	52 x 24	8	1,872	9,984
Desalination Building	Steel Structure	52 x 24	8	1,872	9,984

FIG. 6.3-3. Plot plan of SMART

Design Requirements

For the site-dependent internal or external hazards, the design requirements will be similar to those of advanced evolutionary reactors, in particular with respect to:

- Earthquake,
- Aircraft crash,
- Explosion pressure wave,
- Internal hazards,
- Radiation protection aspects such as accessibility, shielding, ventilation, etc.

6.3.7.2 Reactor building

The reactor building essentially coincides with the containment building, which consists of the concrete shield building and steel containment including the internal structures. Figure 6.3-4 shows a schematic view of the reactor building including part of safety systems. Containment building encloses the safety vessel to confine the release of radioactive products to the outside environment under the postulated beyond design basis accidents relating to the loss of integrity of the safety vessel.

The auxiliary systems for the fuel handling and the spent fuel storage are located in the reactor building.

6.3.7.3 Containment

The safety vessel is a leak-tight pressure-retaining steel vessel intended for the accommodation of all primary reactor systems including the reactor assembly, pressurizer gas cylinders, and associated valves and pipings.

The primary function of the safety vessel is to contain the radioactive fission products within the vessel and thus to protect any primary coolant leakage to the containment. Containment building is therefore required only to possess limited confinement function in case of safety vessel integrity failure or beyond design basis accident.

6.3.7.4 Turbine building

In addition to the normal turbine building characteristics, steam-supplying function to the desalination facility is added. But this additional facility does not influence on the overall size or layout of turbine building.

The criteria of turbine building arrangement include to establish the turbine orientation to consider that the safety related facilities are not impaired by the turbine originated missiles, and to minimize the length of the main steam/feedwater line.

6.3.7.5 Other buildings

The buildings and systems are arranged to optimize the relation between the systems, separation of safety system equipment, maintenance, etc. Other important buildings include auxiliary building, utility building, and desalination facility.

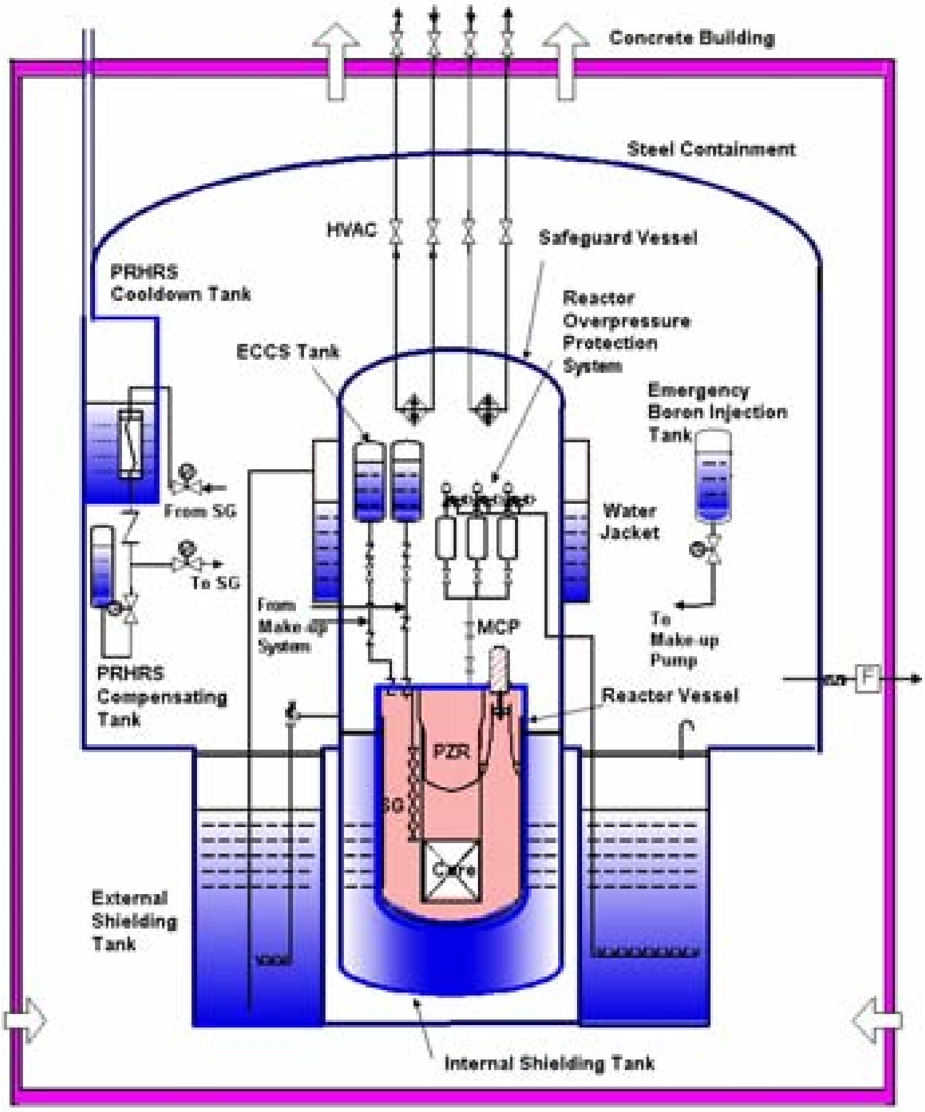

FIG. 6.3-4. Schematic diagram of SMART reactor building

6.3.8 Technical data

General plant data

Power plant output, gross	90	MWe
and 40,000 tons of fresh water /day		
Power plant output, net		MWe
Reactor thermal output	330	MWt
Power plant efficiency, net		%
Cooling water temperature		°C

Nuclear steam supply system

Number of coolant loops		
Primary circuit volume, including pressuriser	56.27	m^3
Steam flow rate at normal conditions	152.7	kg/s
Feedwater flow rate at nominal conditions	152.5	kg/s
Steam temperature/pressure	≥274/3.0	°C/MPa
Feedwater temperature/pressure	180.0/5.2	°C/MPa

Reactor coolant system

Primary coolant flow rate	1550	kg/s
Reactor operating pressure	15	MPa
Coolant inlet temperature, at RPV inlet	270	°C
Coolant outlet temperature, at RPV outlet	310	°C
Mean temperature rise across core	40	°C

Reactor core

Active core height	2.0	m
Equivalent core diameter	1.832	m
Heat transfer surface in the core	821.3	m^2
Fuel inventory	12.47	t U
Average linear heat rate	11.9	kW/m
Average fuel power density	26.462	kW/kg U
Average core power density (volumetric)	62.60	kW/1
Thermal heat flux	401.78	kW/m^2
Enthalpy rise, F_H		
Fuel material	Sintered UO_2	
Fuel assembly total length	2400	㎜
Rod array	square, 17×17	
Number of fuel assemblies	57	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	25	
Number of structural spacer grids	5	
Number of intermediate flow mixing grids		
Enrichment (range) of first core	4.95	Wt% U-235
Enrichment of reload fuel at equilibrium core		
Operating cycle length (fuel cycle length)	36	months
Average discharge burnup of fuel (nominal)	26160	MWd/t
Cladding tube material	Zircaloy	
Cladding tube wall thickness	0.64	㎜
Outer diameter of fuel rods	9.5	㎜
Overall weight of assembly	370.84	kg
Active length of fuel rods	2189	㎜
Burnable absorber, strategy/material	Al_2O_3-B_4C burnable absorber, Gd_2O_3-UO_2	
Number of control rods	49	
Absorber rods per control assembly	24	
Absorber material	Ag-In-Cd	
Drive mechanism	Linear pulse motor	
Positioning rate [in steps/min or mm/s]	4	mm/pulse
Soluble neutron absorber	Boric acid	

Reactor pressure vessel

Cylindrical shell inner diameter		4072	㎜
Wall thickness of cylindrical shell		264	㎜
Total height		10600	㎜
Base material:	Central cover	Stainless steel	
	RPV	Carbon steel	
	liner		
Design pressure/temperature		17/ 350	MPa/°C
Transport weight (lower part), and			t

RPV head		t

Steam generators

Type	Helically-coiled Once-through	
Number [Thermal capacity 27.5 MWt]	12	
Heat transfer surface	168.8	m^2
Number of heat exchanger tubes [per SG]	324	
Tube dimensions	9/12	㎜
Maximum outer diameter	753	㎜
Total height	4522	㎜
Transport weight	2	t
Shell and tube sheet material	PT-3V	
Tube material	PT-7M	

Reactor coolant pump

Type	Canned motor, Axial	
Number	4	
Design pressure/temperature	17/350	MPa/℃
Design flow rate (at operating conditions)	392.3	kg/s
Pump head	17.5	m
Power demand at coupling, cold/hot		kW
Pump casing material	STS321	
Pump speed	3600/900	rpm

Pressuriser

Total volume	24.1	m^3
Steam& gas volume: full power/zero power	12.2	m^3
Design pressure/temperature	17/350	MPa/℃
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter	2408	㎜
Total height	4585	㎜
Material	Stainless steel	
Transport weight		t

Pressuriser relief tank

Not applicable

Total volume		m^3
Design pressure/temperature		MPa/℃
Inner diameter (vessel)		㎜
Total height		㎜
Material		
Transport weight		t

Primary containment

Type	Pressurized concrete with steel lining	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)		m
Free volume		m^3
Design pressure/temperature (DBEs)	0.3/120	MPa/℃
(severe accident situations)		kPa/℃
Design leakage rate		vol%/day
Is secondary containment provided?	No	

Safeguard vessel

Type	Carbon steel with stainless steel coating	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)		m
Free volume	400	m^3
Design pressure/temperature (DBEs)	3/250	MPa/℃
Design leakage rate		vol%/day

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
filter type			
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer, rated voltage			kV
	rated capacity		MVA
Plant transformers,	rated voltage		MVA
	rated capacity		kV
Start-up transformer	rated voltage		kV
	raged capacity		MVA
Medium voltage busbars(6kV or 10kV)			
Number of low voltage busbar system			
Standby diesel generating units: number			
	rated power		MW
Number of diesel-blacked busbar systems			
Voltage level of these			V ac
Number of battery-backed busbar systems			
Voltage level of these			V dc
Number of battery-backed busbar systems			
Voltage level of these			Vac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	Tandem-compound	
Number of turbine sections per unit		
Turbine speed	1800	rpm
Overall length of turbine unit		
Overall width of turbine unit		
HP inlet pressure/temperature	3.0/274	MPa/℃

Generator

Type	3-phase, synchronous	
Rated power		MVA
Active power	100	MW
Voltage		kV
Frequency	60	Hz
Total generator mass [1,216,000 lbs]		t
Overall length of generator		m

Condenser

Type		
Number of tubes		
Heat transfer area		m²
Cooling water flow rate		m³
Cooling water temperature	30	℃
Condenser pressure		kPa

Condensate pumps

Number	2	
Flow rate	151	kg/s
Pump head		m
Temperature	50	℃
Pump speed		rpm

Condensate clean-up system

Full flow/part flow

Filter type

Feedwater tank

Volume		m³
Pressure/temperature		MPa/℃

Feedwater pumps

Number	3	
Flow rate	152.7/3	kg/s
Pump head		
Feedwater temperature	180	℃
Pump speed		rpm

6.3.9 Measures to improve economy and maintainability

6.3.9.1 Summary of measures taken to simplify the design, to reduce costs, construction schedule

Major economic improving features for SMART can be summarized as system simplification, component modularization, factory fabrication & direct site installation of components, reduced construction time.

Integral arrangement of the primary reactor systems requires only a single pressurized vessel and removes large-sized pipes connecting primary components. The adoption of the simplified passive systems provides a net reduction in the number of safety systems, and also drastically reduces the number of valves, pumps, wirings and cables, pipes, etc.

A simplified modular design approach is applied to all SMART primary components. Optimized and modularized small-sized components allow the easy factory fabrication and the direct installation at the site, and thus lead to shortening the construction time and schedule. These features allow a construction period of less than three (3) years from first concrete to fuel load. The compact and integral primary system also eliminates the complexity and extra components associated with the conventional loop-type reactors.

There are some other features contributing to economic improvements. SMART uses advanced on-line digital monitoring and protection system that provides significant advantages of increasing the system availability and operational flexibility. The adoption of advanced man-machine interface technology leads to the reduction of human errors and to a compact and effective design of the control room with respect to minimizing the staff requirements.

The availability factor of the SMART plant is 95%, and the occurrence of unplanned automatic scram events is less than one per year.

6.3.9.2 Summary of measures to achieve high availability and flexibility of operation, and to improve the ability to perform maintenance

In order to improve the reliability and productivity of nuclear plants, it is necessary to perform maintenance on systems and components, whose off-normal functioning can lead to an off-normal operating condition, before serious plant consequences can occur.

However, the basic concept to improve the ability to perform maintenance in SMART is to eliminate the complicated active systems which required periodic observation and maintenance. SMART has great advantages to reduce the maintenances during operation and outages by eliminating the complicated active systems, such as the Chemical Volume and Control System, and pressurizer control system.

Soluble boron-free design contributes to the system simplification by allowing the removal of associated systems and components required for boric acid processing, chemical volume and control systems. This feature also minimizes the liquid radwaste generation and thus simplifies the associated processing systems.

Self controlled pressurizer doesn't need any maintenance while much complicated maintenance is required on the pressurizer spray system and heaters in a conventional commercial power plant.

SMART is designed for a sixty (60) year life and for a three (3) year fuel cycle with a single or one-anda-half batch refueling scheme. The longer refueling period will reduce the number of refueling operations, and the radiation exposure to operating personnel.

The possibility of coolant leaking through joints between reactor vessel and MCP, CEDM and central cover, was reduced by adopting a welded torus sealing.

6.3.10 Project status and planned schedule

The SMART development program was launched in November 1996. Before this program started, there was an R&D project which was focused on the investigation of new and innovative elementary technologies and various studies on the concept of the advanced reactor systems. Based on the results of this previous R&D, SMART was decided to be an integral type PWR producing a rated thermal power of 330 MW.

The program of the conceptual study was then extended to the conceptual design program from July 1997, as one of national mid- and long-term nuclear R&D programs. Comprehensive efforts had been pursued on the safety enhancement of the SMART concept by implementing passive safety design features and also by utilizing the physically inherent safety characteristics. Along with the conceptual design, simultaneous efforts had been put on the development of various technologies including design methodology, computer codes, fundamental experiments and tests, etc. The conceptual design was then completed in March 1999. The preliminary safety analyses on the selected limiting design bases events were carried out to assure the safety of the concept of the SMART system. From these results, it was shown that the SMART system was properly responding to those events by meeting the safety criteria.

The successful completion of the conceptual design for the SMART led to the three year-term basic design phase, which was begun in April 1999. The basic design of the SMART reactor system has been completed in March of 2002. The six-year long 2nd phase project of the SMART development was launched in July 2002. The objectives of this project are to verify the integral performance of SMART system technologies and to confirm the industrial applications through the construction and operation of a 1/5 scaled-down pilot plant, SMART-P. The SMART-P project will be carried out by a consortium of the government and domestic nuclear industries with the strong support from the nuclear society.

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6.4 SSBWR (HITACHI LTD., JAPAN)

6.4.1 Introduction

A safe and simplified BWR (SSBWR) has been developed as an innovative future reactor to provide a super-long life core of 20 years and to realize a passive core safety system with infinite grace period. Operability and maintainability can be largely improved by using the super-long life core, reducing the number of active components, and using a one-batch core with no exchange of fuel assemblies, which is also a feature that provides proliferation resistance. The reference output powers are 150MWe and 434 MWth, suitable for a distributed small energy source. Design targets of the SSBWR are set as follows.

(1) Simplicity of Operation and Maintenance
- 20-year core life
- One-batch core
- Reduced number of active components
- Reduced amount of high level radioactive waste
- Proliferation resistance

(2) Safety and Reliability
- Core insensitive to pressure transient at accidents
- Naturally started safety system
- Infinite grace period
- Elimination of large LOCAs

(3) Multi-purpose Energy Supply
- Indirect cycle BWR

(4) Cost Reduction
- Reduced number of components
- Compact design of RPV, PCV and BOP systems

Long life operation of the core can be achieved by using a less moderated neutron spectrum by selecting the fuel lattice, fuel material and/or coolant. An indirect cycle is used for flexibility in co-generation applications using high temperature secondary steam. An infinite grace period can be achieved using the heat pipe system with external heat sinks.

6.4.2 Description of the nuclear systems

6.4.2.1 Primary circuit and its main characteristics

Fig. 6.4-1 shows the Reactor Pressure Vessel (RPV) and reactor internals. There is no liquid/vapour separation system above the core in the RPV, and steam generators (SGs) are inserted into the steam region so that a compact indirect cycle can be applied using the high temperature secondary loop for a multi-purpose energy source. Natural recirculation is used for core cooling, which eliminates recirculation pumps. During normal operation, primary water is heated and evaporated in the core and the steam is cooled and condensed at the steam generator. Primary water is operated in the saturated condition where the RPV is self-pressurized and there is no additional pressurizer. Secondary pressure is maintained at 7.1 MPa to keep the thermal efficiency the same as that of the BWR, but primary pressure is increased to 12.3 MPa to reduce SGs volume under the minimum critical power ratio (MCPR) condition. The RPV with 4.2 m diameter and 15.2 m height is made compact by eliminating pumps and using small SGs with high condensation heat transfer. There are no feed water or no primary coolant steam lines so that the probability of a large break LOCA is greatly reduced.

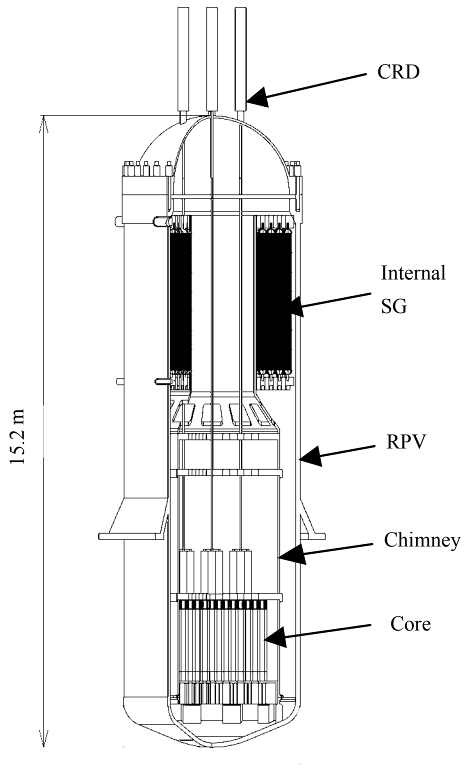

FIG. 6.4-1. The SSBWR reactor pressure vessel

6.4.2.2 Reactor core and fuel design

Super-long life operation of 20 years can be achieved with a less moderated neutron spectrum by using heavy water as a coolant and a triangular tight fuel lattice in core fuel assemblies of enriched UO_2 (Fig. 6.4-2). A less moderated neutron spectrum improves the internal conversion ratio which reduces burnup reactivity. This less moderated neutron spectrum also introduces a small dependency of core reactivity on coolant void fraction which realizes mild transient behavior and small reactivity swing between hot and cold state as shown in Fig. 6.4-2. There are two kinds of fuel assemblies. The first type of fuel assembly (Type I) has no guide thimbles for cluster control rod insertion and consists of 397 enriched UO_2 fuel rods. The second type of fuel assembly (Type II) has 55 guide thimbles and 342 fuel rods. There are no channel boxes in each type of fuel assemblies. Coolant flow is separated by thin plates set at the core support as a honeycomb. Fuel assemblies are loaded between those plates. There are 7 control rod drivers; 6 control rod arms are branched from one control rod drive and each arm has 55 cluster control rods which are inserted into the Type II fuel assembly. Gadolinium is mixed in 6 fuel rods of each fuel assembly type to maintain sub-criticality in case light water is injected into the core. The core consists of 91 inner core fuel assemblies of 12.9wt% UO_2 and of 60 outer core ones of 18.1%wt% UO_2. The equivalent core diameter is about 3.4m and the core height is 1.2m. Discharge fuel burnup is about 68 GWd/t. The core can decrease large excess reactivity due to its super long life and one-batch core operation and it can decrease the reactivity swing between normal operation and cold shutdown states by using heavy water and supplemental burnable poison. The excess reactivity is about 7.8% Δ k/k. Increase of coolant void reactivity due to core burnup can be compensated by diluting heavy water with light water gradually. Initial primary coolant is just heavy water; light water is gradually mixed with heavy water to become 70% of the coolant after 16 years. The small excess reactivity can reduce the necessary number of CRDs and simplified RPV internals and lower cost can be achieved.

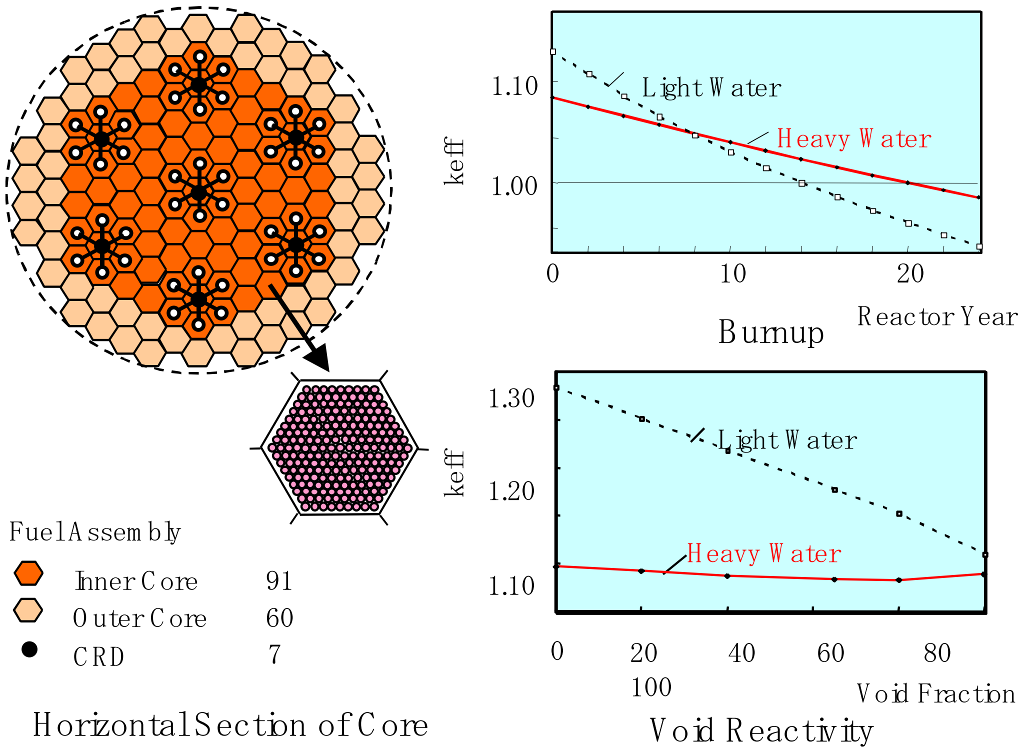

FIG. 6.4-2. The SSBWR long life core

There is also a potential to reduce the long-term radioactivity hazard from high level waste because the SSBWR core has a harder neutron spectrum suitable for transmutation of minor actinides (MAs). Transmutation of Np-237 was evaluated because a large amount of Np-237 is confined in waste fuel and its half-life is quite long (2.14×10^6 year). The rate of decrease of reactivity with burnup is smaller with more Np-237 because the neutron capture cross section of Np-237 is larger than that of U-238, and the fission rate of daughter nuclei such as Pu-238 increases during burnup. Void reactivity increases with the amount of Np-237 and the mixing rate of Np-237 is limited at 5wt% under the restriction of void reactivity less than 5$. The total amount of transmuted Np-237 is almost the same as the amount of yearly products from three 1000MWe LWRs or from 20 SSBWRs. Np-237 can also be transmuted in thermal reactors, but the amount of Np-237 which can be loaded into the core is limited due to its large capture cross section. In the SSBWR, however, Pu-238 is fissile and the burnup characteristics are much better than those of thermal reactors because the neutron spectrum of the SSBWR is very hard like that of a fast breeder reactor (FBR).

6.4.2.3 Fuel handling and transfer systems

The core life is 20 years without re-fueling and there is no need for fuel handling systems. The fuel handling building, therefore, is eliminated. After the operation, the core is removed from the RPV by a crane and transferred to a core storage tank located over the PCV. The core can be transferred, being kept inside the RPV, for further restriction of nuclear proliferation resistance.

6.4.2.4 Primary components

Reactor pressure vessel

The RPV is a cylindrical welded vessel with a top head. There are seven nozzles on the top head to insert the CRDs in the direction of gravity. There is no large nozzle for primary water supply lines and main steam lines on the RPV because the primary water is enclosed in the RPV. There are four sets of nozzles for secondary water supply lines and steam lines. There are also small nozzles for the CUW (clean-up water) line and for safety systems. All nozzles are located over the core outlet so that the core is covered with water even in the case of LOCA accidents induced by pipe breaks. The shroud support is attached to the lower barrel and fixes the core support panel.

Reactor internals

The core is fixed on the core support and surrounded by the shroud on the shroud support. The shroud is attached to the upper core panel and supports the chimney over the core. The chimney, which enhances natural circulation, is cylindrical and its diameter is decreased about the water level so that the SGs are installed surrounding the chimney in the steam region above the water level. Water is separated gravitationally over the core in the chimney and dry steam can flow into the SGs so that high condensation heat transfers are kept at the SGs.

For the ABWR, the thermal power can be controlled by flow rate of primary coolant using primary pumps, but there is no primary pump for the SSBWR. Thermal power, therefore, can be changed by the CRD system. The chimney has a lot of holes and the natural recirculation flow rate is changed linearly with the water level where opening rate is ～2%. Flow rate can be decreased by opening the holes, but natural recirculation capability is sufficient to cool down the core with a chimney of 5m height. MCPR is evaluated as over 1.3 for normal operation. Natural recirculation is maintained by using the holed chimney, even if the water level is lowered in a LOCA.

Steam generator

The internal SGs are located above the core and steam generated in the core is condensed. Condensed water is returned to the core through an annular down comer outside the chimney. The SGs are shell-tube type heat exchangers with the shell side having primary coolant and the tube side having secondary coolant. The secondary coolant flows into the steam turbine and the outlet temperature of the secondary coolant is 560K at 7.1 MPa, which is almost the same as the ABWR. The SGs are compact due to high heat transfer of condensation at the outer wall of the tube and boiling at the inner wall of the tube. The SGs are also used to remove decay heat from the core with natural convection of primary coolant.

Pressurizer

Primary water is operated in the saturated condition where the RPV is self-pressurized and there is no additional pressurizer.

Reactor coolant pumps

Natural recirculation is adopted for core cooling, and reactor coolant pumps are eliminated.

Main coolant lines

The primary system of the SSBWR reactor is of the integral type which includes internal SGs and confines primary coolant in the pressure vessel. Therefore, there are no main coolant feed water or main steam lines.

6.4.2.5 Reactor auxiliary systems

Decay heat can be removed by the SGs and CUW heat exchangers. Decay heat is removed by the internal SGs using the secondary coolant system, but the SGs are located in the steam region and primary coolant can not be cooled down by the SGs under no boiling condition. CUW heat exchangers can be used for cooling down primary water under saturated temperature.

Chemical and volume control system

Water chemistry of the primary loop is controlled by the CUW line to maintain the quality of water and to remove corrosion products and fission products. Heavy water is diluted by light water to suppress the increase of void reactivity with burn-up and the dilution rate is controlled by the CUW line.

Hydrogen is generated due to the radiation in the core and flows into the steam. Hydrogen and oxygen are re-combined to water by a catalyst on the reactor internals in the steam region. Tritium generated from deuterium due to the radiation, therefore, is included in water and removed in the CUW line.

The CUW line is also used for controlling the water level in the RPV.

6.4.2.6 Operating characteristics

During normal operation, the reactor power is mainly varied by the control rod driving system and partially by the control system of the water level in the RPV. The turbine control system is also connected to the system controlling reactor power. The secondary coolant is a multi-purpose energy source for the heat utilization system. The heat and power generation ratio can be controlled due to the consumer demand, and reactor power can be operated at any level between 50% and 100%.

During start-up operation, non-condensed gas is evacuated from the RPV in order to achieve high condensation at the SGs, and the core starts to heat up as control rods are drawn up. The reactivity swing can be compensated by the control rod worth.

6.4.3 Description of turbine generator plant systems

An indirect cycle is applied and the turbine generator plant is isolated from the primary coolant. Radiation shields are greatly reduced in comparison with that of the conventional BWR plant.

6.4.3.1 Turbine generator plant

The steam cycle is a single step reheat cycle with moisture separator and reheaters. Steam generated in the SGs flows through four steam pipes, isolation valves, a steam mixing header and a pressure control valve before it reaches the steam generator. Discharged steam is fed into a condenser.

6.4.3.2 Condensate and feedwater systems

Condensed water in the condenser is pressurized by a low-pressure pump. The feed water line is a single line with feed water heaters, three for low pressure and one for high pressure.

The clean-up system for secondary coolant has a condensate demineralizer to maintain water quality and to prevent the accumulation of deposits onto structures.

6.4.3.3. Auxiliary systems

Secondary coolant flow is also fed into the heat utilization system. High temperature steam is used through intermediate heat exchangers for multi-purpose energy source such as industrial process heat source, air conditioning and heating source and so on.

6.4.4 Instrumentation and control systems

6.4.4.1 Design concept, including control room

The reactor control system includes the reactor power control system, reactor pressure control system and reactor water level control system, which keep main parameters such as power, pressure, water level within a proper range during changes and disturbances in normal operational conditions and power load.

For a modular reactor, control rooms for each reactor are combined into a single control room where automatically controlled systems are operated by fewer operators. Operability and operation cost can be significantly improved.

6.4.4.2 Reactor protection and other safety systems

The reactor protection system includes the reactor emergency shutdown system and engineered safety features actuation system. The system is triggered by plural and independent outputs of pressure, temperature, flow rate, water level, gamma/neutron flux and valve position gauges. The reactor protection system activates mainly passive systems such as gravitational scram and automatically opening valves with electricity breaks.

6.4.5 Electrical systems

6.4.5.1 Operational power supply systems

A power generator is connected to a steam turbine and the generator capacity is 167500 KVA for a 150 MWe plant. Electricity on site is provided by the generator's own output power and a diesel generator is installed for reactor start-up and also for emergency use. Electricity can be provided during start-up, normal and accident operations. However, electricity is not necessary for terminating the accident because passive systems are used.

6.4.5.2 Safety-related systems

Safety systems are mostly passive but decay heat can be removed mainly by secondary flows using the SGs. A diesel generator is operated to supply secondary flow by pumps for a total blackout condition. Even if the diesel generator is not operated, passive systems such as ADS and GDCS are activated to cool down the core.

6.4.6 Safety concept

6.4.6.1 Safety requirements and design philosophy

Safety requirements are (1) the prevention of accidents, (2) the reliable termination of accidents and (3) the mitigation of accidents, which are basically the same as those of the conventional reactors, but such requirements are achieved by new concepts.

For the prevention of accidents, the SSBWR is designed so that the possibilities of accidents are reduced or eliminated. The core has small void reactivity and insensitive response to pressure transient. Natural circulation of primary coolant assures core cooling even in the case of total blackout. The possibilities of large break LOCA are greatly reduced by using the integral type reactor without large nozzles for primary coolant on the RPV.

For termination of accidents, multiple and independent systems of CRD and a boron water injection system are installed.

For the mitigation of accidents, the ECCSs are activated to cool down the core. Reliability can be enhanced by naturally induced passive safety systems that eliminate active components such as pumps. Decay heat can be removed by heat pipes, which are naturally cooled down by the outer air even when the SGs are out of order. The grace period is expected to be almost infinite because the decay heat is removed by natural convection while maintaining the water inventory, and evacuation of the public is not expected to be necessary.

Deterministic design basis

The SSBWR is a combination of the BWR and the PWR reactor types because it is an indirect cycle boiling water reactor. The categories of events should be taken into account for both types of reactors but many events were eliminated by using a one-batch core, an integrated natural circulation primary cooling system and passive safety systems. Events like refueling accidents and accidents related to feed water and steam lines such as loss of flow accidents and main steam line break are eliminated.

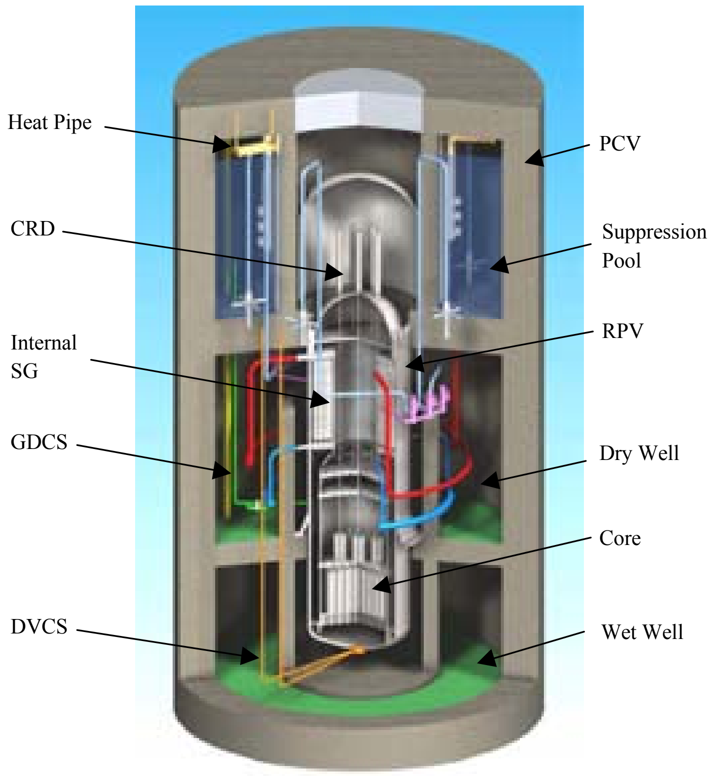

FIG. 6.4-3. The SSBWR primary containment vessel

Loss of coolant accidents

There are no large feed water pipes or primary coolant steam lines that are connected to the RPV. Therefore, the phenomena investigated for LOCAs are just small piping breaks such as the CUW line. The ECCSs are equipped with the automatic depressurized system (ADS) and the gravitational driven coolant injection system (GDCS), which are designed to achieve the conditions of no core uncovery and no fuel cladding heatup during LOCAs. After the system pressure decreases with ADS operation, the GDCS is activated.

Fig. 6.4-3 shows the Primary Containment Vessel (PCV) and safety systems of the SSBWR. The PCV is divided into three stages, the upper suppression pool, the middle dry well and the lower wet well. The gas region above the upper suppression pool is connected to the lower wet well by vent pipes. Here, the wet well at the bottom of the PCV is a closed confined room where non-condensed gas is contained. In the small LOCA, pressure increases in the PCV and non-condensed gas initially confined in the dry well flows into the suppression pool. Then discharged steam in the dry well enters the suppression pool where steam is condensed. The temperature of the suppression pool is increased and evaporation starts to occur. Non-condensed gas goes into the wet well through vent pipes and steam fills in the gas region above the suppression pool. Heat pipes which connect the gas region above the suppression pool and the outer space of the PCV can remove heat from within the PCV effectively due to high heat transfer of condensation in the gas region above the suppression pool. Decay heat gradually balances with cooling of heat pipes and SGs inside the RPV. Once the GDCS is activated, steam is generated at the core, passed through the dry well, suppression pool and GDCS piping before being returned to the core. Water, therefore, effectively circulates in the PCV to cool down the core.

In a typical PCV, the pressure is the sum of non-condensed gas pressure and steam saturated pressure. On the other hand, for the SSBWR, PCV pressure is equal to the larger of the two, the non-condensed gas pressure or the steam pressure because non-condensed gas goes to the wet well. The PCV pressure of the SSBWR, therefore, is less than that of a typical PCV for equivalent accident conditions. The thin PCV wall cuts construction cost without compromising safety. According to the thermal hydraulic calculation for a small LOCA with a 100% break of CUW piping attached to the RPV, the two-phase level is higher than the top of the core and core uncovery does not occur; therefore heatup of the fuel cladding also does not occur during the transient. The heat pipe can be activated to transfer heat in the PCV to outer space and decay heat can be removed with an infinite grace period.

SG accidents

For SG tube break accidents, the concerned SGs are isolated and the reactor can be partially operated by using the remaining SGs. For large break accidents of SGs, isolation valves of secondary loop for all SGs are closed and ADS+GDCS are activated.

Risk reduction

The core damage frequency will be less than 10^{-5}/year with intrinsic and passive safety systems although evaluation of probability figures for core damage has not been completed because new safety systems are still to be included into the conceptual design.

External and internal hazards

There are no CRD systems and no recirculating pumps in the lower part and the core is located near the bottom of the reactor building. Those features are advantageous for seismic design. The seismic level for the BOP system is reduced in comparison with the conventional BWR due to the adoption of the indirect cycle. The PCV, which is a reinforced concrete containment vessel (2m thick), is tougher in comparison with ship-shell structured vessel and provides advantages against external flying objects.

6.4.6.2 *Safety systems and features (active, passive and inherent)*

Safety systems configuration

For reactor shutdown, CRD and boronated water injection system are installed. The CRs are inserted gravitationally into the core and boron water is injected into the primary coolant from an accumulation tank. Decay heat can be removed by recirculating secondary coolant using pumps. For the ECCS, water is injected gravitationally to cover the core and decay heat can be removed by a naturally started heat pipe system which is effective even in the case SGs are unavailable. The safety systems include active and passive systems, but those are multiple systems backed up by passive systems.

Safety injection systems

The GDCS and the direct vessel cooling system (DVCS) are installed to provide water inside the RPV to cover the core and outside the RPV to assure the in-vessel retention (IVR).

In-containment refueling water storage tank

There is a core storage tank but no refueling water storage tank because of the adoption of a one-batch core.

Emergency feed water system

There is no emergency feed water system because the primary coolant is naturally recirculated and confined in the RPV.

Residual heat removal system

Decay heat can be removed normally by the SGs and the heat exchangers of the CUW. For accidents, decay heat can be also removed by heat pipes which go through the PCV.

6.4.6.3 *Severe accidents (beyond design basis accidents)*

Severe accident mitigation strategy

The probability of severe accidents is very low, but the accidents are considered to terminate safely without failure. The IVR should be achieved by water injection systems to prevent further expansion of accidents and prevent leakage of radioactive materials outside the PCV.

Severe accident prevention and mitigation features

Even if the isolation valve of the GDCS is not activated, the water level in the RPV can be maintained by condensation of internal SGs. But, if both isolation valves of the GDCS and internal SGs are not activated, the water level is gradually lowered in the RPV and the core might be exposed. In this case, the core can melt down because there is not enough cooling. Core debris is piled up on the lower inner wall of the RPV and it heats up the RPV wall so that the debris might penetrate the RPV and flow out. To prevent this situation, fused valves connected to the suppression pool by the DVCS pipes are installed in contact with the RPV lower wall. Even if there is core melt down, the fused valves are opened by the temperature rise of the RPV heated by the melted debris and the lower part of the RPV is filled with water without any damage. Steam generated at the lower dry well surrounding the lower part of RPV circulates through the dry well, DVCS pipes and lower dry well. Condensed water in the dry well is drained into the lower dry well. Water, therefore, circulates in the PCV effectively to cool down the core for the severe accident as well and the IVR is achieved.

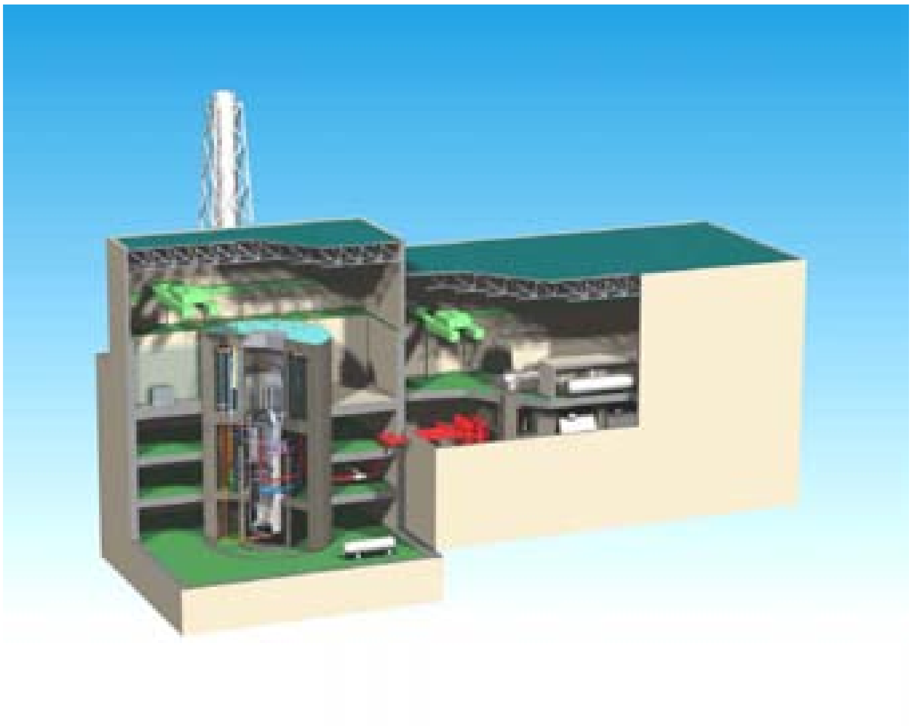

FIG. 6.4-4. The SSBWR reactor & turbine buildings

6.4.7 Plant layout

6.4.7.1 Buildings and structures, including plot plan

The reactor and turbine buildings of a single modular plant are shown in Fig. 6.4-4. Those buildings are separated for maintenance and seismic treatment because radioactivity and seismic levels of the turbine system are lower than those of the reactor system due to the indirect cycle. The I-type layout of those buildings is applied for easy extension to multiple modular plants.

Seismic Behaviour and Design requirements

Design requirements are as follows.

- Layout for high safety and easy maintenance
- Layout for reduced radiation exposure of employees and the public
- Layout for tolerance of geographical conditions including earthquake frequency
- Layout for reduced cost

The reactor building including the core should be seismic resistant but the turbine system is almost the same as that of the fossil power plant.

6.4.7.2 Reactor building

The reactor building is 31 m wide and 34 m high, including the RPV, the Primary Containment Vessel (PCV), the ECCS, CUW system, and other systems related to the primary system. The cylindrical PCV is located at the centre of the building in the basement so that the PCV and the building are formed as a single structure.

The CUW system including pumps, heat exchangers and water tanks are located in the lower part of the building. There is no CRD system and recirculating pumps in the lower part and the core is located near the bottom of the building. Those configurations are advantageous for seismic design.

Decay heat in an accident is removed by heat pipes installed at the top of the PCV and there are chimneys 30 m high to enhance air cooling of the heat pipes.

6.4.7.3 Containment

The PCV is a three staged compact reinforced concrete containment vessel (RCCV) of the pressure suppression type. The three-staged configuration of the PCV makes pressure equal to the larger of the non-condensed gas pressure or the steam pressure because non-condensed gas goes to the lower wet well. The PCV pressure, therefore, is less than that of a typical PCV. The RCCV is surrounded by reinforced concrete which is advantageous for seismic and shield design. The PCV is accessible for entry from the top and there are also side hatches for maintenance.

6.4.7.4 Turbine building

The turbine building includes the turbines, condenser, feed water pumps for secondary coolant, condensed water clean-up system and systems related to secondary coolant. A sea-water cooling system including heat exchangers and pumps is also installed.

6.4.7.5 Other buildings

There are no additional buildings because the systems usually housed in an auxiliary building are included in the reactor building and a fuel handing building is not necessary because of the one-batch core.

6.4.8 Technical data

General plant data		
Power plant output, gross	150	MWe
Power plant output, net		MWe
Reactor thermal output	434	MWt
Power plant efficiency, net	34.5	%
Cooling water temperature		℃
Nuclear steam supply system		
Number of coolant loops	1	
Primary circuit volume, including pressurizer		m3
Steam flow rate at normal conditions	234	kg/s
Feedwater flow rate at nominal conditions	234	kg/s
Steam temperature/pressure	287/7.1	℃/MPa
Feedwater temperature/pressure	216/7.1	℃/MPa
Reactor coolant system		
Primary coolant flow rate	1600	kg/s
Reactor operating pressure	12.3	MPa
Coolant inlet temperature, at RPV inlet	327	℃
Coolant outlet temperature, at RPV outlet		℃
Mean temperature rise across core		℃
Reactor core		
Active core height	1.2	m
Equivalent core diameter	3.8	m
Heat transfer surface in the core		m2
Fuel inventory		tU
Average linear heat rate	6.3	kW/m
Average fuel power density		lW/kgU
Average core power density (volumetric)	40	kW/l
Thermal heat flux, Fq		kW/m2

Enthalpy rise, FH			
Fuel material			UO2
Fuel assembly total length			mm
Rod array		triangle, hexagonal	
Number of fuel assemblies		151	
Number of fuel rods/assembly(Type I/II)		397/342	
Number of control rod guide tubes		7	
Number of spacers			
Enrichment (range) of first core		15	Wt%
Enrichment of reload fuel at equilibrium core		-	Wt%
Operating cycle length (fuel cycle length)		(240)	months
Average discharge burnup of fuel		68	MWd/t
Cladding tube material			SUS316
Cladding tube wall thickness		0.68	mm
Outer diameter of fuel rods		11.7	mm
Overall weight of assembly			kg
Active length of fuel rods		1200	mm
Burnable absorber, strategy/material			Gd
Number of control rods		7	
Absorber rods per control assembly			
Absorber material			
Drive mechanism			
Positioning rate			
Soluble neutron absorber		-	
Reactor pressure vessel			
Cylindrical			
Shell inner diameter		4.2	m
Wall thickness of cylindrical shell		0.2	m
Total height		15.2	
Base material:	cylindrical shell	low alloy steel	
	RPV head	low alloy steel	
	Liner	SUS	
Design pressure/temperature		14MPa/337℃	

Transport weight (lower part), including head		
RPV head		
<u>Steam generators</u>		
Type	Shell/Tube	
Number	4	
Heat transfer surface	2535	m2
Number of heat exchanger tubes		
Tube dimensions	22/1.3	mm
Maximum outer diameter	3800	mm
Total height	4000	mm
Transport weight		t
Shell and tube sheet material		
Tube material	Inconel 600	
<u>Reactor coolant pump</u>	Not Applicable	
Type		
Number		
Design pressure/temperature		MPa/℃
Design flow rate (at operating conditions)		kg/s
Pump head		m WG
Pump casing material		
Pump speed		rpm
<u>Pressurizer</u>	Not Applicable	
Total volume		m3
Steam volume: full power/zero power		m,3
Design pressure/temperature		MPa/℃
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter		mm
Total height		mm
Material		
Transport weight		t
<u>Pressurizer relief tank</u>	Not Applicable	
Total volume		m3
Design pressure/temperature		MPa/℃
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t
<u>Primary containment</u>		
Type		
Overall form (spherical/cyl.)		cylindrical
Dimensions(diameter/height)	15/26.3	m
Free volume(drywell/wetwell/cod.pool)	1300/930/550m3	
Design pressure/temperature (DBEs)	408/171	kPa/℃
(severe accident situations)	815/171	kPa/℃
Design leakage rate	0.4vol%/day	
Is secondary containment provided?	No	
<u>Reactor auxiliary systems</u>		
Reactor water cleanup, capacity	5.6	kg/s
Filter type	ion exchange resin	
Residual heat removal, at high pressure		kg/s
at low pressure		kg/s
Coolant injection at high pressure		kg/s
at low pressure		kg/s
<u>Power supply systems</u>		
Main transformer, rated voltage		kV
rated capacity		MVA
Plant transformers, rated voltage		kV
rated capacity		MVA
Start-up transformer rated voltage		kV
Rated capacity		MVA
Medium voltage busbars		kV
Number of low voltage busbar systems		V
Standby diesel generating units: number		
rated power		MW
Number of diesel-backed busbar systems		
Voltage level of these		V ac
Number of DC distributions		

Voltage level of these		V ac
Number of battery-backed busbar systems		
Voltage level of these		V ac

Turbine plant

Number of turbines per reactor		
Type of turbine(s)		
Number of turbine sections per unit (e.g. HP/LP/LP)		1HP/2LP
Turbine speed	1500	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	6.92/	MPa/℃

Generator

Type		
Rated power		MVA
Active power		MW
Voltage		kV
Frequency	50	Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type		
Number of tubes		
Heat transfer area		m2
Cooling water flow rate		m3/s
Cooling water temperature		℃
Condenser pressure		hPa

Condensate pumps

Number		
Flow rate		kg/s
Pump head		MPa
Temperature		℃
Pump speed		rpm

Condensate clean-up system

Full flow/part flow		
Filter type		

Feedwater tank

Volume		m3
Pressure/temperature		MPa/℃

Feedwater pumps

Number		
Flow rate	234	kg/s
Pump head		MPa
Feedwater temperature	216	℃
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages		
Redundancies		

6.4.9 Measures to enhance economy and maintainability

The primary cooling system is simplified by eliminating steam separator/dryer system and recirculation pumps. The cost increase due to the heavy water can be compensated by the cost reduction due to the elimination of the number of CRDs. Additional internal S/Gs are inserted in the RPV, but the BOP system is simplified due to reduced seismic and shield design. In comparison with ABWR, ECCSs are simplified by eliminating the high-pressure core flooder (HPCF) and the low pressure flooder (LPFL) with pumps and adding ADS and GDCS without active components. The decay heat removal system is the secondary cooling system of internal S/Gs and the CUW system in cooperation with the RHR system. The compact PCV can be realized by using the heat pipe system to transfer decay heat outside and keep low pressure inside. Driving electric power is very small and load rate on site is lowered and negligibly small because the number of active components is reduced. The capital costs of construction are reduced by eliminating active components and systems can be contained in a small PCV and reactor building. Operation and maintenance costs are dramatically decreased due to the long life operation and the reduction of movable components. SSBWR has the possibility to overcome the inferiority of the small reactor, and total life-cycle costs including construction, operation and maintenance are competitive to those of the large reactor (ABWR).

6.4.10 Project status and planned schedule

The first stage of conceptual design is almost complete and includes innovative concepts related to the following aspects.

- Long-life core of 20 years,
- Compact RPV and reactor internals,
- Passive safety systems and the PCV,
- Compact reactor and turbine buildings.

The next stage is to extend the conceptual design with additional concepts and to verify the design by experimental tests. Material tests and development especially are needed to realize a super long-life core. The reactor without a super long-life core will be deployed in the 2010s but the project with a super long-life core will be completed for a commercialized reactor in the 2020s – 2030s.

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6.5 IMR (MITSUBISHI, JAPAN)

6.5.1 Introduction

A small reactor is more capable of matching the demands for gradual increases in the electric power market, as well as of reducing the risk of investment in Japan. A small reactor also better meets the demand for small markets such as those in developing countries where infrastructures are insufficient. Therefore, small reactors should be favorably developed. However, small reactor designs need developments of innovative technology and modularity to overcome various disadvantages in economical efficiency, inherent to small-scale reactors such as high construction, operation and maintenance costs. In addition, a high level of safety, which is required for all NPPs, can help to assure a high level of reliability and simple operation and maintenance for small reactors.

The Integrated Modular water Reactor (IMR) is a 300MWe class small reactor being developed by Japanese colleagues (Mitsubishi, Kyoto University, the Central Research Institute of the Electric Power Industry (CRIEPI), and the Japan Atomic Power Company) to meet these market demands.

6.5.2 Description of the nuclear systems

6.5.2.1 Primary circuit and its main characteristics

Figure 6.5-1 shows the plant concept of the IMR. The IMR is an integral type water reactor where the primary system components are all installed inside the reactor vessel. Its features are as follows.

1) Main coolant piping and primary coolant pumps are eliminated by adopting integrated natural circulation system.
2) A pressurizer is eliminated by adopting the self-pressurization system.
3) Control rod drive mechanisms (CRDM) are located inside the reactor vessel (RV).
4) There are two types of steam generators (SG) located in the reactor vessel. One is located in the vapor portion in the RV (called SG in vapor (SGV)), and the other is located in the liquid portion in the RV (called SG in liquid (SGL)).
5) Steam generators are also used as decay heat removal heat exchangers during accidents and normal startup and shutdown operations.
6) A passive safety system, which does not require any external support (Stand-alone Direct Heat Removal System: SDHS), is adopted.
7) Emergency core cooling systems are eliminated.

The hybrid heat transport system (HHTS) is employed to transport the fission energy released in the fuel to the steam generators by both vapor formation and liquid temperature rise. The energy transported by vapor produces secondary steam in SGV, and the energy transported by liquid temperature rise produces secondary steam in SGL. The SGV also has a function of primary system pressure control, and the SGL has the function of core power control through the core inlet temperature by controlling the feedwater flow rate.

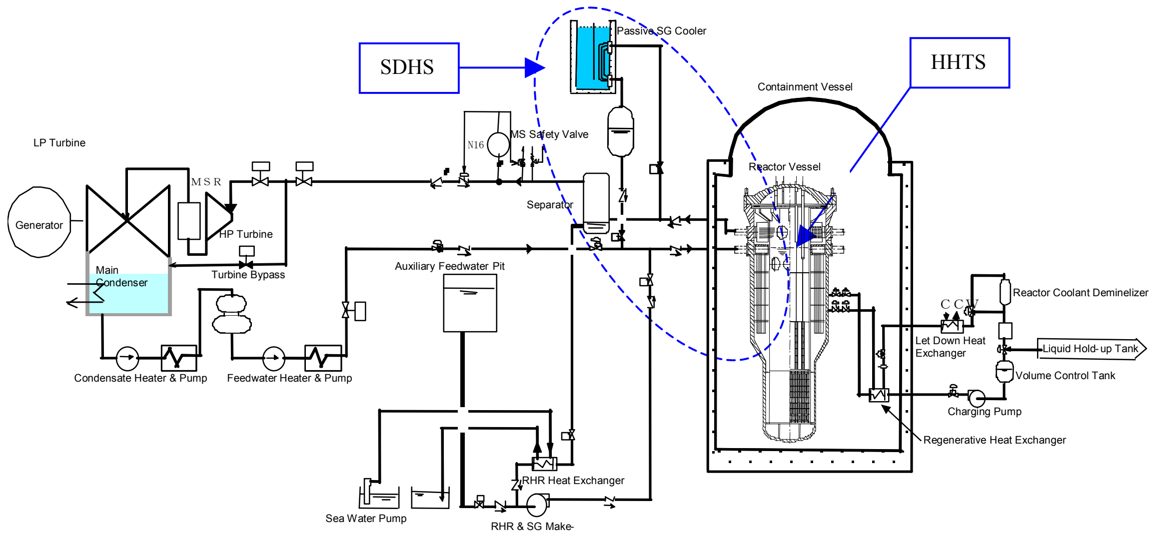


FIG.6.5-.1 Plant concept of IMR

6.5.2.2 Reactor core and fuel design

Figure 6.5-2 shows the cross section of the IMR reactor and Table6.5-I shows the major IMR parameters in comparison with the current conventional PWR. The reactor core, consisting of ninety-seven 21x21 fuel assemblies, has a thermal output of approximately 1,000 MWt. To maintain the core thermal margin and to achieve a long fuel cycle, the core power density is reduced to 1/3-1/2 of a conventional PWR. The design-refueling interval is 3 years in 3 batches of fuel replacement. The fuel rod design is the same as that for a conventional PWR.

The chemical shim reactivity control is not adopted but both control rods which contain enriched boron 10 and burnable absorbers control the whole reactivity.

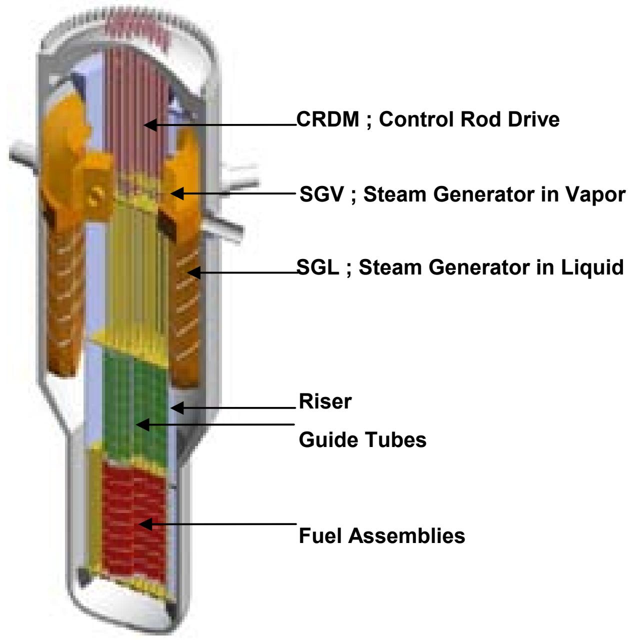


FIG. 6.5-2. Bird View of IMR reactor

TABLE 6.5-I. MAJOR IMR PARAMETERS

	IMR	**Existing four loop PWR**
Electric power output	Approx. 330MWe	1,180 MWe
Thermal power	Approx. 1,000 MWt	3,423 MWt
Fuel assembly type	21x21	17 × 17
Number of fuel assemblies	97	193
Fuel effective length	Approx. 3.7 m	Approx. 3.7 m
Total uranium inventory	Approx. 69 tones U	Approx. 89 tones U
Number of control rods	92	53
Reactor vessel	6 m max. inner dia. & 20 m height	Approx. 4.4 m inner dia. & 13 m height
Steam generators	U type (SGL) and C type (SGV)	52F type
Primary coolant pumps	None	93A-1 type
Primary system flow (kg/h)	10×10^6	60×10^6
Turbine	No information provided.	TC6F44
Containment	RCCV	PCCV
Engineered safety systems	Stand-alone type passive	Two trains
Refueling water storage	Outside containment	Outside containment
Reactor protection system	Digital	Analog
Reactor control system	Digital	Digital
Main control room	Improved	Standard

6.5.2.3 Fuel handling and transfer systems

The fuel handling and fuel transfer systems consist mainly of the refueling crane and the multi-functional mast type spent fuel pit crane which covers reactor cavity and spent fuel pit. Considering the recent need to reduce the periodical inspection time, many improvements including an increase in the speed of each system have been introduced. Also, in order to reduce operations in radiation controlled areas; these systems can be operated fully-automatically by one operator from a remote centralized control room instead of the present method of operating individually from a control station next to each piece of equipment.

6.5.2.4 Primary components

Reactor pressure vessel

The upper part of the reactor vessel's inside diameter is about 6 m in order to accommodate the in-vessel steam generators. The inside diameter of the lower part of the reactor vessel is reduced to about 4 m in order to reduce the cold-side water volume. In order to eliminate the necessity for the consideration of LOCA, the largest diameter nozzle connected to the reactor vessel is reduced to less than 10 mm. In addition, the lowest location of the nozzle is above the core to improve the reliability of the reactor vessel.

Reactor internals

The core is located in the bottom of the reactor vessel and the steam generators are located in the upper part of the reactor vessel. Control rod guide assemblies are located above the core and a riser is set above the control rod guide assemblies to enhance the natural circulation.

Two types of in-vessel CRDMs are adopted. One is motor driven CRDM, which is applied to the control bank. The basic development of this CRDM has been conducted by the Japan Atomic Power Research Institute (JAERI). This CRDM has the function of controlling reactivity during operation by finely stepping the control rod position. The other is the hydraulic type CRDM. This CRDM has the scram function and applied to the shut-down bank. The control rods connected to this CRDM are moved by hydraulic force from the bottom position to the top, and then held by electro-magnetic force. When the scram signal is initiated, the control rods are released and inserted into the core by gravity by turning off the power to the CRDM.

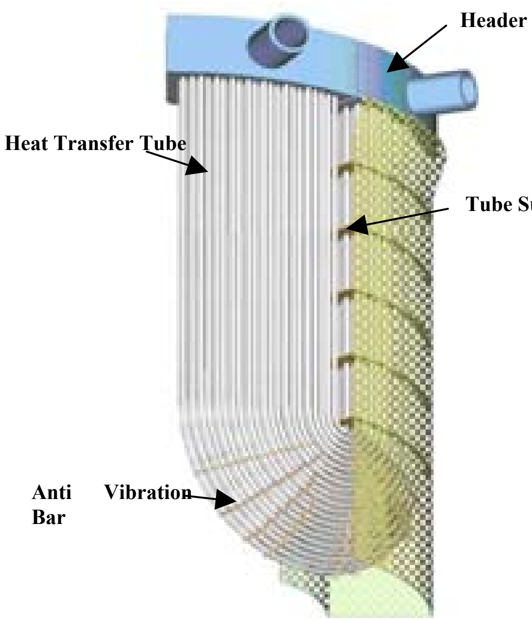
Steam generators

There are two types of steam generators adopted in IMR. One is the steam generator in vapor (SGV), which is located above the water level in the reactor vessel. The energy transported by vapor formation generates secondary steam through SGV. In addition, since the vapor in the reactor vessel is condensed by SGV, controlling the feedwater flow rate to SGV can control the reactor vessel pressure.

The other is the steam generator in liquid (SGL), which are located in the water in the reactor vessel. The energy transported by liquid temperature rise generates secondary steam through SGL, and also, since the core inlet temperature can be controlled by the amount of heat removal through SGL, the core power can be controlled by controlling the feedwater flow rate to SGL. By this method, the movement of the control rods for controlling reactor power will be minimized.

Figure 6.5-3 shows SGL. A U-type tube bundle is adopted, since it is necessary to minimize pressure drops on both the primary and secondary sides to maintain good natural circulation performance.

A C-type steam generator is adopted for SGV as shown in Figure 6.5-4, because of the good space utilization in the vapor part of the reactor vessel.



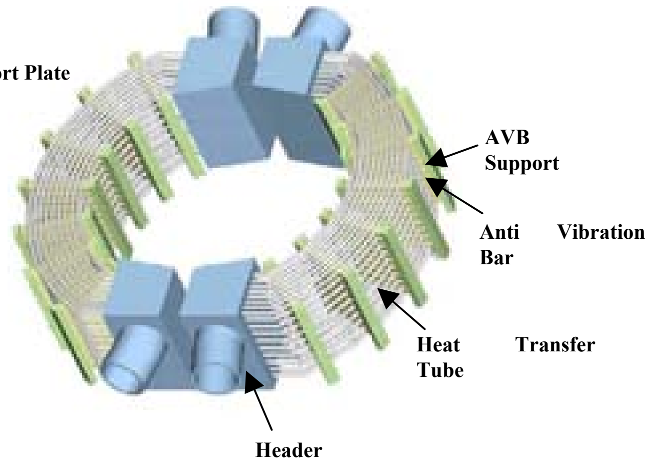


FIG.6.5-3. SG in Liquid (SGL) *FIG.6.5-4. SG in Vapor (SGV)*

Pressurizer

The pressurizer is eliminated by adopting the self-pressuriztion system.

Reactor coolant pumps

Reactor coolant pumps are eliminated by adopting the two-phase natural circulation system.

Main coolant lines

Main coolant pipes are eliminated by adopting the integral type reactor vessel.

6.5.2.5 Reactor auxiliary systems

Chemical and volume control system

The chemical and volume control system has the following two main functions:

The first function is to adjust the amount of water contained in the reactor vessel. In normal operation, the letdown and charging flows are controlled so that the water level in the reactor vessel is maintained at the programmed level.

The second function is to purify the primary coolant. A demineralizer and filter in the letdown line purify the primary coolant. Hydrogen gas will be added in the volume control tank to minimize the oxygen concentration in the primary coolant.

The letdown flow is taken from the side of the reactor vessel, and the coolant is cooled by the regenerative heat exchanger and letdown heat exchanger, and then purified in the demineralizer. To supply water to the primary coolant system, two charging pumps are used to extract water from the volume control tank.

Reactivity control system

Controlling feedwater flow rate to SGL will mainly perform the power control. In addition, the control rods will be used to assist. The absorber material of the control rods have B10 enriched B_4C. Burnable absorbers mainly compensate the reactivity change accompanied with fuel burn-up with assistance of control rods. Gd_2O_3 integrated in fuel pallets and separate absorber rods containing boron are applied as burnable absorber.

6.5.2.6 Operating characteristics

The reactor is designed so that it can be operated automatically within the range of 20 to 100% of rated output by the reactor control system. Even in the low output range below 20%, the control system can control the reactor automatically in the low power-operating mode.

Usually, the primary system pressure and reactor power are controlled feedwater and control rods. When the load to the plant is changed, the feedwater flow rate and control rods are controlled simultaneously to steadily maintain the plant.

The reactor control system is designed so that it can allow the following load demand change without causing a reactor trip:

- A +/- 10% step load demand change (within the range of 20 to 100%)
- A +/- 5% per min ramp load demand change (within the range of 20 to 100%)
- 100% load rejection

6.5.3 Description of the turbine generator plant systems

6.5.3.1 Turbine generator plant

No information provided.

6.5.3.2 Condensate and feed water systems

No information provided.

6.5.3.3 Auxiliary systems

No information provided.

6.5.4 Instrumentation and control systems

6.5.4.1 Design concept, including control room

No information provided.

6.5.4.2 Reactor protection and other safety systems

No information provided.

6.5.5 Electrical systems

6.5.5.1 Operational power supply systems

No information provided.

6.5.5.2 *Safety related systems*

No information provided.

6.5.6 Safety concept

6.5.6.1 *Safety requirements and design philosophy*

Significant improvements have been achieved in the safety goals and safety philosophy of the IMR. By adopting an integral type primary system, accidents that may cause fuel failure, such as loss of coolant accidents (LOCA), rod ejection (R/E), loss-of-flow (LOF) and locked rotor (L/R), are eliminated. Since the diameter of the pipes connected to the primary system (reactor vessel) is limited to less than 10 mm, the water level in the reactor vessel can be maintained at normal levels by water injection from the charging pumps.

For the mitigation system, the Stand-alone Direct Heat Removal System (SDHS) is achieved. Figure 6.5-5 shows the concept of SDHS. In this system, the decay heat is removed directly from inside the reactor vessel to the atmosphere. Therefore, even if water leakage occurred and the charging pumps did not work, water leakage would be terminated automatically when the pressures inside and outside the reactor vessel are equalized. In the Passive Steam Generator Cooler (PSGC), decay heat is removed by water-cooling in the early stage of the accident and then, the heat transfer mode is gradually replaced by air-cooling. Therefore, external support such as water, power and operators are not necessary for maintaining plant safety.

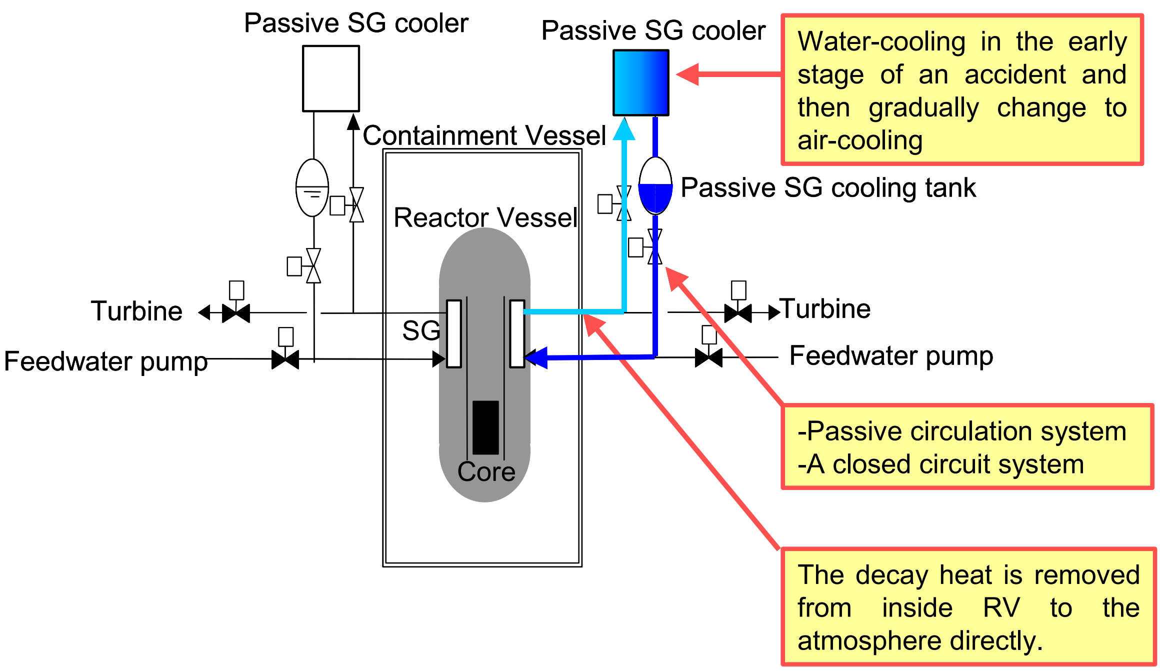


FIG. 6.5-5. Stand-alone Direct Heat Removal System

Deterministic design basis

Design basis events are abnormal operating conditions, which are classified into two groups: abnormal operating transients and accidents during operation, and safety criteria have been set for each group. The standards for radiation exposure are specified for normal operation and accidents, thus reducing the risk to the general public and employees to less than an allowable limit.

As shown before, design basis accidents, which may cause fuel damage, have been completely eliminated in the IMR.

Risk reduction

To further reduce the risk and provide increased protection, the reactor is designed to have a high degree of safety with simplification, economy, and ease of maintenance taken into account. Significant risk reduction is accomplished in the IMR as follows:

- No fuel damage is expected during design basis accidents.
- No external support is required to maintain the plant in a safe condition after accidents.
- A passive safety system with an infinite grace period is adopted.
- Reactor vessel integrity will be retained even if severe accidents occur.
- Containment vessel integrity will be maintained by submerging the containment vessel head through operator action, even if severe accidents occur.

External and internal hazards

No information provided.

6.5.6.2 Safety systems and features (active, passive and inherent)

Safety systems configuration

There are two trains of the Stand-alone Direct Heat removal System (SDHS). Therefore, if a malfunction such as steam generator tube leakage were to occur, system functions would be maintained. Since the nozzle diameter connected to the primary system is limited to less than 10 mm, the safety injection function of the safety systems is eliminated. The capacity of chemical and volume control system (CVCS) is covered with eight 3/4-inch pipes which is connected to reactor vessel because of the small reactor coolant inventory.

Safety injection systems

Safety injection systems were eliminated by adopting the SDHS and by limiting the nozzle diameter connected to the primary system.

Containment spray system

The containment spray system was eliminated by adopting SDHS.

Auxiliary feedwater system

The auxiliary feedwater system is used for startup and shutdown procedures during normal operation. It is also used at the initial stage of the transients caused by the secondary systems. When the plant safety is threatened only by the auxiliary feedwater system, the safety system (SDHS) will actuate. The auxiliary feedwater system is not a safety system, because the plant safety is guaranteed by SDHS.

Residual heat removal system

The auxiliary feedwater system is also used as the residual decay heat removal system.

6.5.6.3 Severe accidents (beyond design basis accidents)

Basic concept

Figure 6.5-6 shows the measures for severe accidents. When decay heat removal through SGs is not applicable, water leaked out of the reactor vessel will submerge to the bottom of the reactor vessel cavity. Since the decay heat can be removed through the reactor vessel wall, molten fuel will be retained inside the reactor vessel. In addition, decay heat in the containment vessel will be removed through the containment head, which will be immersed in the water supplied by operators.

Countermeasures during plant shutdowns

Residual heat will be removed through steam generators during plant shutdown. When steam generators are replaced for inspection, fuel assemblies in the reactor vessel will be removed before the steam generator replacement.

Mitigation of severe accidents

Since there are two trains of passive safety systems, the possibility of loss of the decay heat removal path is very small. In addition, water can be injected into the reactor vessel by charging pumps and decay heat can be removed through the containment vessel head.

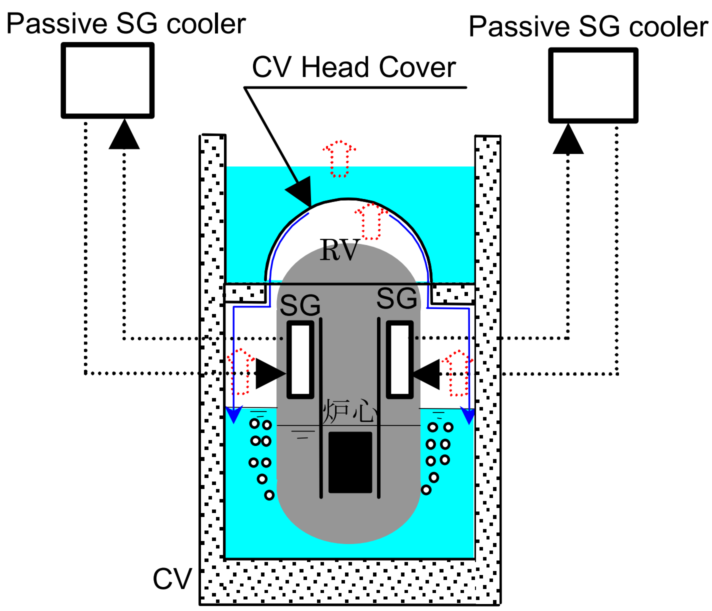


FIG. 6.5-6. Measures for severe accidents

6.5.7 Plant layout

(1) Design conditions

1) Containment

A reinforced concrete containment is used. The design pressure of the containment is planned to be 9 MPa to meet the safety requirement that water leakage from reactor vessel will be terminated automatically by equalling pressure inside and outside the reactor vessel. Since this containment vessel is about one size larger than the reactor vessel, it will be able to resist high pressure.

2) Reactor building

Ground level is assumed at above sea level, and flat land is assumed. The bedrock is assumed to be less than 40 meters below ground to enable the use of pile foundations. The reactor building is adopted in an isolated building and the turbine building isn't adopted. The reactor building's configuration is as follows:

- Integrated building consisting of two units
- Exclusion of waste disposal facilities in another building
- Adoption of steel structures in the building considering the module method

3) Fuel handling facilities

Based on the current PWR concept, the fuel handling facilities are assumed to have the following facilities or areas:

- Fuel exchange cavity
- Core internal storage pit
- Spent fuel pit (SFP) and accessories
- Laydown space on the operation floor
- Fuel handling building (FHB) crane and spent fuel pit crane
- Cask trailer area

(2) Plant design

1) Containment

The containment wall has the double role of being the walls of the reactor building. A cylindrical containment is chosen, because the shape can withstand pressure easily. The inside dimension of the containment is inevitably decided by the size of the reactor vessel.

2) Reactor building

Figure 6.5-7 shows the layout of the reactor building on the operation floor. The plant facilities and areas, such as the SFP crane, the FHB crane, core internal storage area, containment cover, and so forth, are shared between units. The steel plate reinforced concrete structure is applied to the radiation control area for the requirement of shielding. The hull structure is applied to the non-radiation area without the shielding requirement.

6.5.7.1 Buildings and structures, including plot plan

For the economic efficiency, the plant design has the following goals;

- Compact layout of the reactor building,
- Reduction of construction bulk, and
- Shortening site work periods by the module method.

For these goals, specific items are selected as follows;

- Share of plant facilities among units,
- Reactor building including containment,
- Isolated devices, and
- Adoption of steel structures for the reactor building.

Figure 6.5-7 shows the general arrangement of the plant. This figure shows an example of the twin-unit type.

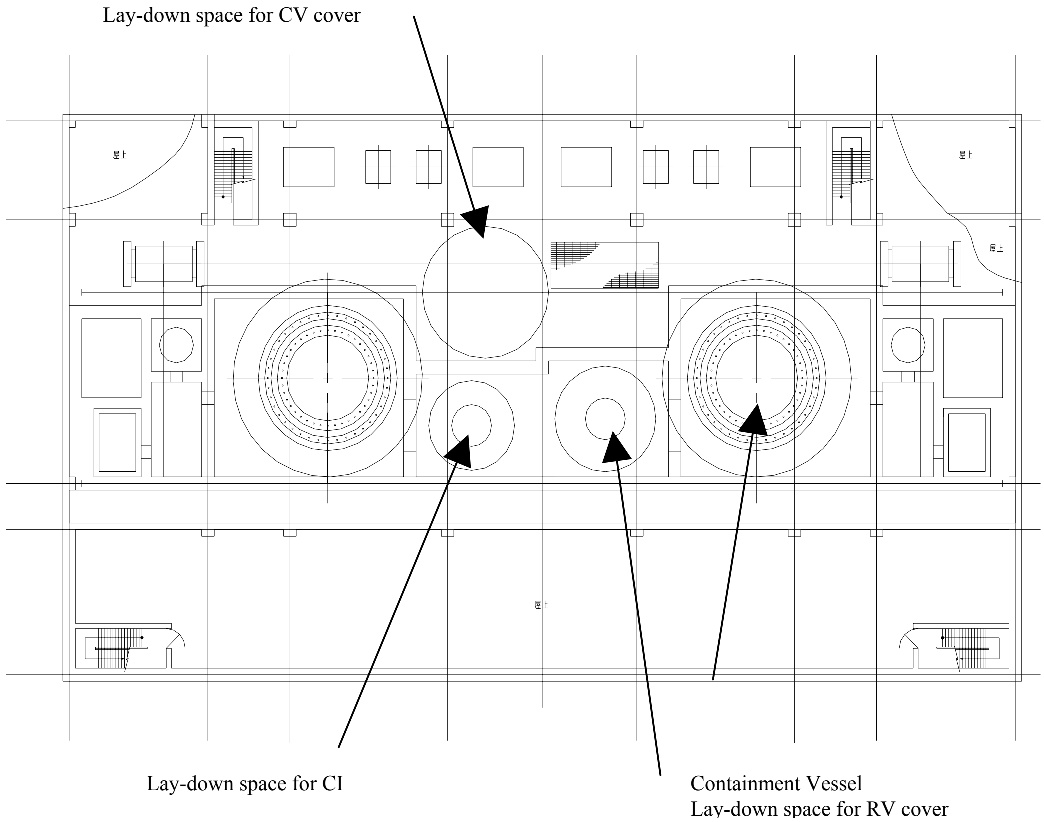


FIG. 6.5-7. IMR General arrangements

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 into the environment if an accident occurs, and those, which have serious consequences for the plant.

Class B: Those which have relatively small 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 as Class A, the waste building as Class B, and the turbine building as 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 the 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.

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 designed so 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 in such a way that the exposure dose to the general public 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.

6.5.7.2 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 6.5-8 shows a cross-section of the reactor building.

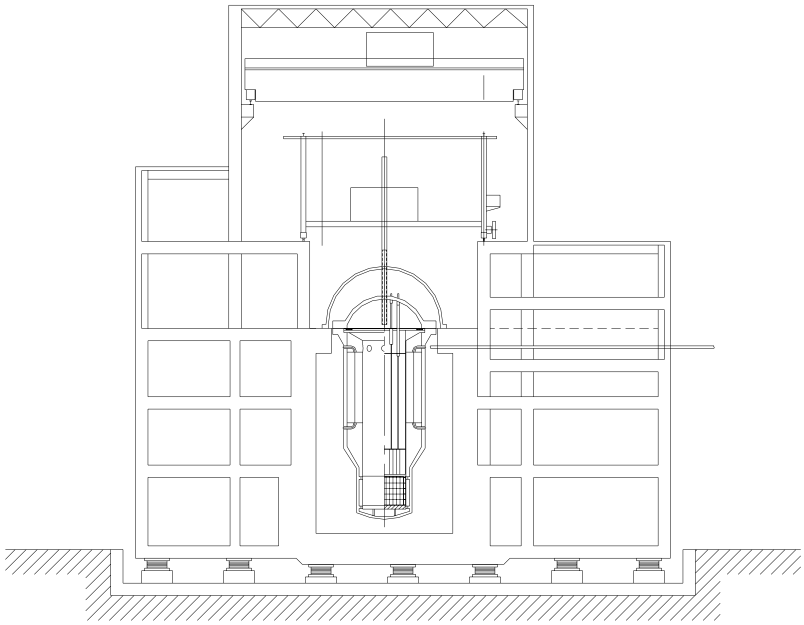


FIG. 6.5-8. IMR Axial cross section

6.5.7.3 Containment

The containment is part of the reactor containment facility. The reactor containment facility is part of the engineered safety systems, which include SDHS.

The containment system is designed to suppress or prevent the possible dispersion of 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 feed water system.

6.5.7.4 Turbine building

The turbine generator, turbine, condenser, moisture separator and reheater (MSR) and their auxiliary equipments are installed in the turbine building. The turbine generator is arranged with its axis in line with the reactor.

Suitable spaces have been provided for inspection access, the transportation of tools for inspections and maintenance, and disassembly in such a way that the volume of the building is reduced.

6.5.7.5 Other buildings

The buildings and systems have been arranged so as to optimize the relation between the systems, the separation of safety system equipment, seismic resistance, maintenance, etc.

Fuel handling building

The fuel handling building is the part of the reactor building, which contains the spent fuel pool, the cask pit, the fuel inspection pit and associated equipments.

Control building

The control building, which is common to both units, contains mainly the main control room, electrical equipment and access control equipment.

Waste building

The waste building, which is common to both units, mainly houses the radioactive waste treatment systems.

6.5.8 Technical data

General plant data

Power plant output, gross	330	MWe
Power plant output, net		MWe
Reactor thermal output	1,000	MWt
Power plant efficiency, net	Approx. 35	%
Cooling water temperature		°C

Nuclear steam supply system

Number of coolant loop	none	
Primary circuit volume	Approx. 250 m^3	
Steam flow rate at nominal conditions	510	kg/s
Feedwater flow rate at nominal conditions	510	kg/s
Steam temperature/pressure	290 / 5	°C/MPa
Feed water temperature/pressure	220 / 5	°C/MPa

Reactor coolant system

Primary coolant flow rate	3000	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	307	°C
Coolant outlet temperature, at RPV outlet	345	°C
Mean temperature rise across core	38	°C

Reactor core

Active core height	3.7	m
Equivalent core diameter	3.0	m
Heat transfer surface in the core	8.7	m^2
Fuel inventory	62	tU
Average linear heat rate	7	kW/m
Average fuel power density	16	kW/kgU
Average core power density (volumetric)	42	kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		
Fuel material	Sintered UO_2	
Fuel assembly total length	4,000	mm
Rod array	square, 21x21	
Number of fuel assemblies	97	
Number of fuel rods/assembly	385	
Number of control rod guide tubes	32/assembly	
Number of spacers		
Enrichment (range) of first core	less than 5	wt%
Enrichment of reload fuel at equilibrium core	less than 5	wt%
Operating cycle length (fuel cycle length)	3	years
Average discharge burnup of fuel	45,000	MWd/t
Cladding tube material	Zr 4	
Cladding tube wall thickness	0.5	mm
Outer diameter of fuel rods	9	mm
Overall weight of assembly		mm
Overall weight of assembly		kg
Active length of fuel rods	Approx. 3700	mm
Burnable absorber, strategy/material	Integrated Gd fuel and separable B rods	
Number of control rods assemblies (RRC)	92	
Number of grey control rods assemblies (GRC)	none	
Number of water displacer rods assemblies (WDR)	none	
Absorber rods per control assembly	32	
Absorber material:	RCC enriched B_4C	
Drive mechanism:	Motor driven and hydraulic	
Positioning rate		steps/min [or mm/s]
Soluble neutron absorber	none	

Reactor pressure vessel

Cylindrical shell inner diameter	max.6, 000	mm
Wall thickness of cylindrical shell	270	mm
Total height	max. 20,000	mm
Base material: cylindrical shell	low alloy steel	
RPV head	low alloy steel	
Design pressure/temperature	/	MPa/°C
Transport weight (lower part)		t
RPV head		t

Steam generators in liquid (SGL)

Type	U type	
Number	4	
Heat transfer surface	1,900	m^2/SGL
Number of heat exchanger tubes	4,000	/SGL
Tube dimensions	11	mm
Maximum outer diameter		mm
Total height	8,000	mm
Transport weight		t
Shell and tube sheet material	TT 690 alloy	
Tube material	TT 690 alloy	

Steam generators in Vapor (SGV)

Type	C type	
Number	2	
Heat transfer surface	760	m^2/SGV
Number of heat exchanger tubes	3,500	/SGV
Tube dimensions	11	mm
Maximum outer diameter		mm
Total height	1,500	mm
Transport weight		t
Shell and tube sheet material	TT 690 alloy	
Tube material	TT 690 alloy	

Reactor coolant pump

Type		
Number	none	
Design pressure/temperature		MPa/°C
Design flow rate (at operating conditions)		kg/s
Pump head		
Power demand at coupling, cold/hot		kW
Pump casing material		
Pump speed		rpm

Pressuriser none

Total volume		m^3
Steam volume: full power/zero power		m^3
Design pressure/temperature	/	MPa/°C
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter		mm
Total height		mm
Material		
Transport weight		t

Pressuriser relief tank

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Dry	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)		m
Free volume		m^3
Design pressure/temperature (DBEs)	/	kPa/°C
(Severe accident situations)	/	kPa/°C
Design leakage rate	<0.1	vol%/day
Is secondary containment provided?		

Reactor auxiliary systems

Reactor water cleanup,	capacity	kg/s
	Filter type	
Residual heat removal,	at high-pressure	kg/s
	At low pressure	kg/s
Coolant injection,	at high-pressure	kg/s
	At low pressure	kg/s

Power supply systems

Main transformer,	rated voltage	kV

	Rated capacity		MVA
Plant transformers,	rated voltage		kV
	Rated capacity		MVA
Start-up transformer	rated voltage		kV
	Rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)		6.6	kV
Number of low voltage busbar systems			
Standby diesel generating units: number			
	Rated power		MW
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)		
Number of turbine sections per unit (e.g. HP/LP/LP)		
Turbine speed		rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/°C

Generator

Type	
Rated power	MVA
Active power	MW
Voltage	kV
Frequency	Hz
Total generator mass	t
Overall length of generator	m

Condenser

Type	
Number of tubes	
Heat transfer area	
Cooling water flow rate	m^2
Cooling water temperature	m^3/s
Condenser pressure	hPa

Condensate pumps

Number	
Flow rate	kg/s
Pump head	
Temperature	°C
Pump speed	rpm

Condensate clean-up system

Full flow/part flow	
Filter type	

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/°C

Feedwater pump

Number	
Flow rate	kg/s
Pump head	
Feedwater temperature	°C
Pump speed	rpm

Condensate and feedwater heaters

Number of heating stages	
Redundancies	

6.5.9 Measures to enhance economy and maintainability

The followings show a summary of measures to enhance the economics and maintainability.

To simplify the design

The followings are the IMR features to simplify the design compared with conventional PWRs.

- Natural circulation system eliminates reactor coolant pumps (RCPs) and main coolant pipes.
- Self-pressurization system eliminates the pressurizer.
- Stand-alone direct heat removal system (SDHS) eliminates safety injection system, containment spray system, emergency core cooling system and accumulator tanks.
- Hybrid heat transport system eliminates Boric-related system and reduces waste disposal system.

To improve the operation flexibility

No information is provided.

To reduce the cost of equipment and structures

No information is provided.

To reduce the construction period

IMR applies the module method, which reduces the quantity of the on-site work, to reduce the construction period. Since the module block has building structures without concrete, and equipments, pipes, cables, ducts, and their supports, the quantity of on-site work will be greatly reduced.

To reduce the scope of maintenance during operation and outage

Since the number of equipments or safety systems are greatly reduced compared with conventional PWRs, there are few maintenance items for safety equipments. In addition, since the design-refueling interval is 3 years in 3 batches of fuel replacement by adopting low power density core, work-load for refueling will be greatly reduced.

To make the maintenance easier and with lower radiation exposure

No information is provided.

To increase the plant availability and load factor

Since the design-refueling interval is 3 years in 3 batches of fuel replacement by adopting low power density core and the refueling is the critical terms in the periodic inspection, the plant availability is increased.

To reduce the power generation cost

Our Target will overcome the disadvantages of scale by simplification of systems explained above and series unit construction.

6.5.10 Project status and planned schedule

No information provided.

6.6 KLT-40 FLOATING NUCLEAR HEAT AND POWER UNIT WITH KLT-40S REACTOR PLANT (OKBM, RUSSIAN FEDERATION)

6.6.1 Introduction

Russia North regions and similar remote regions altogether occupy about 50 per cent of Russia's territory, and about twenty million people live there. Remoteness from water transport routes which are useable the year-round, and from railways, is characteristic of these areas. At the same time abundant mineral resources are discovered and exploited there. Two-thirds of Russia's natural resources are located in the Russian North regions, and implementation of these resources requires considerable power capacity. The United Power System of Russia covers only 15% of the country's territory; thus the northern regions of Russia are located in a decentralized power supply zone, where small power sources prevail.

Development of local nuclear power sources with relatively low capacity is a subject of vital importance for the regions, and it is a part of the general strategy for Russia's nuclear power development within the framework of the "Power Units for Light and Heat" Project.

Small-power plants for various purposes represent a specific kind of nuclear power, characterized by compact configuration, enhanced safety and environmental compatibility.

This line of nuclear power is capable of satisfying various users of heat and electricity. The nuclear power plants are preferable compared to other power sources at least in 11 most prospective points of the Russian North-East.

The idea to develop and supply a floating power unit (FPU) constructed at a shipyard, tested and handed over on a turnkey basis in an industrially developed region at minimum construction and mounting activity at a site is very attractive both from technical and economical points of view.

Important arguments in favor of a floating nuclear co-generating plant based on FPU against alternative fossil-fired sources of comparable capacity include the relatively small scope of construction activity at a site under northern conditions (where the environment is vulnerable to antropogenic (man-induced) impacts), the high ecological friendliness and, if necessary, the possibility to relocate a plant to another site.

At present, the activity on a lead nuclear co-generation plant design, which is based upon a floating power unit with a KLT-40S reactor plant (developed by OKBM as a modified version of the commercial KLT-40-type propulsion plants for the icebreaker fleet in Russia), is the most advanced.

To date, the Basic Design of the reactor plant has been developed and approved, design activities on a floating power unit are coming to completion, the site for the plant location has been selected and licensing issues are currently solved with GAN (the Russia regulatory body).

A floating nuclear co-generating plant, based on the floating power unit with reliable and safe reactor plant, verified by long operation will ensure a possibility for the most rapid elimination of the power and heat deficit in northern regions through construction of floating power units at shipyard works for shortest terms at minimal costs simultaneously with a site preparation.

These advantages are real backgrounds to start with the implementation of a floating nuclear co-generating plant based on FPU with KLT-40S reactor plant as a power source for North regions and other similar Russia regions.

Russia is the only country in the world possessing a powerful nuclear icebreaking-transport fleet, which is called upon for performance of important social-economic tasks of northern regions of the country, providing a year-round navigation through the Arctic sea route.

A single nuclear transport & power generating complex, consisting of nuclear-powered ships and floating heat-and power co-generating stations, based on FPU with standard reactor plant serviced by a common service infrastructure, will provide an effective solution of many problems specific for Russia North and similar regions at minimum expenditures.

A general view of the Small Power Nuclear Co-Generation Plant (SP NCGP) is given in Fig. 6.6-1, and some main technical and economic indices are given in Table 6.6-I.

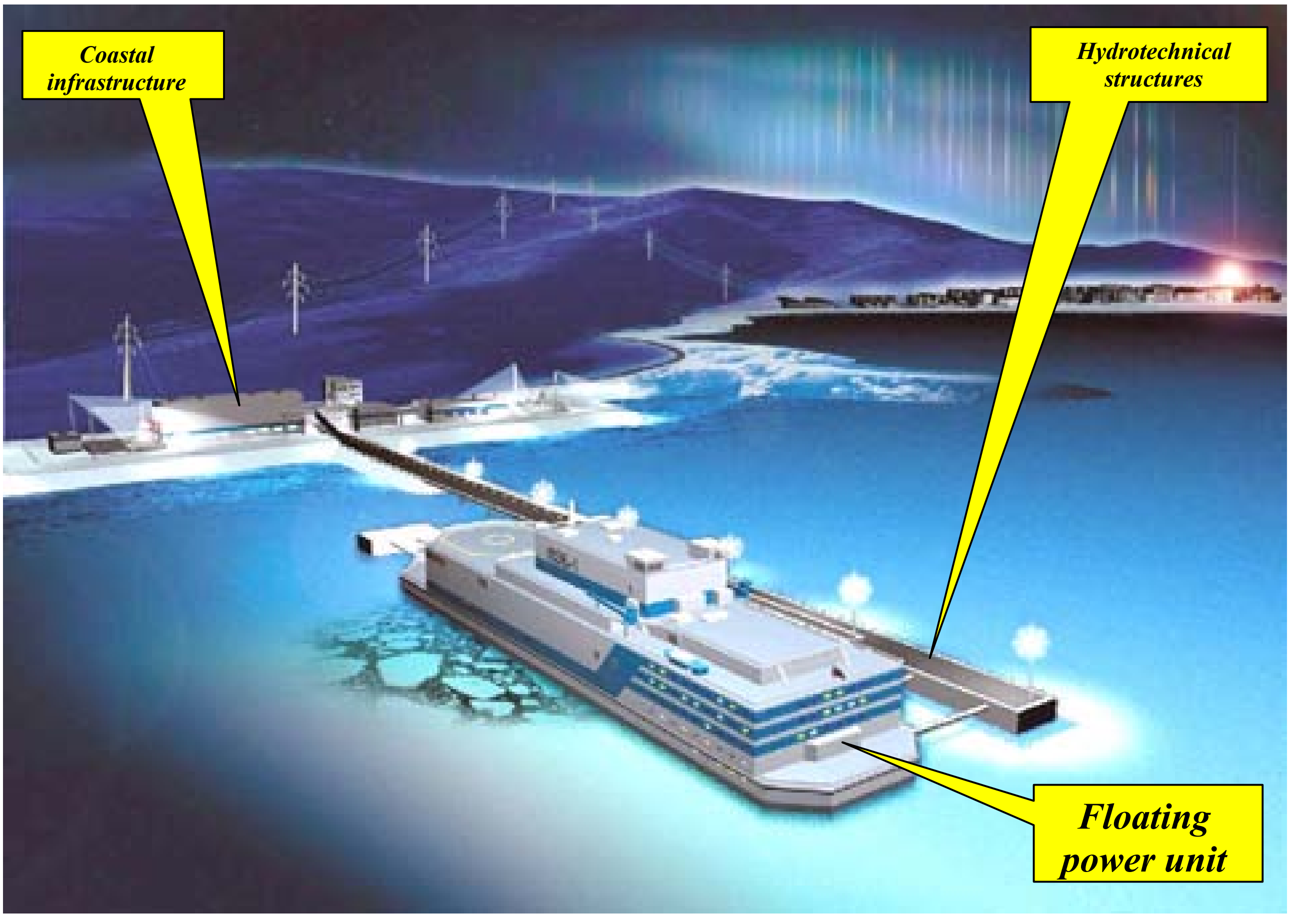


FIG. 6.6- 1. General view of SP NCGP.

TABLE 6.6-I. MAIN TECHNICAL AND ECONOMICAL INDICES OF A SMALL POWER NUCLEAR CO-GENERATION PLANT ON THE BASIS OF FPU WITH THE KLT-40S REACTOR PLANT.

Characteristic	Value
Electric power transmitted to the Consumer, MW	up to 70
Thermal power transmitted to the Consumer, Gcal/hr	up to 140
Electric energy manufacturing cost, $/kW·hr	0,05
Thermal power manufacturing cost, $/Gcal	4,8
Service life, years	40

6.6.2 Description of the nuclear systems

The KLT-40S reactor plant is a complex of systems and elements intended to convert nuclear energy into thermal energy. It includes a nuclear reactor and components directly connected to it, which are needed for normal operation and safety provision.

Major rated technical parameters

- thermal capacity	150 MW
- steam-generating capacity	240 t/h
- primary circuit pressure	12.7 MPa
- steam pressure at SG exit	3.72 MPa
- superheated steam temperature	290 ^{0}C

Reliability parameters

- service life	(35-40) years
- overhaul period	(10-12) years
- non-removable equipment lifetime	(240-300)·10^3 h
- removable equipment lifetime	(80-100)·10 h
- continuous operation period	8000 h.

Figures 6.6-2 and 6.6-3 show the general view of the reactor plant. The KLT-40S reactor plant is a two-circuit plant with a PWR. The main reactor plant (RP) equipment includes a reactor, steam generator and pumps structurally connected by main pipes and forms a steam generating block. (see Fig. 6.6-4)

FIG. 6.6-2. General view of KLT-40S RP reactor room.

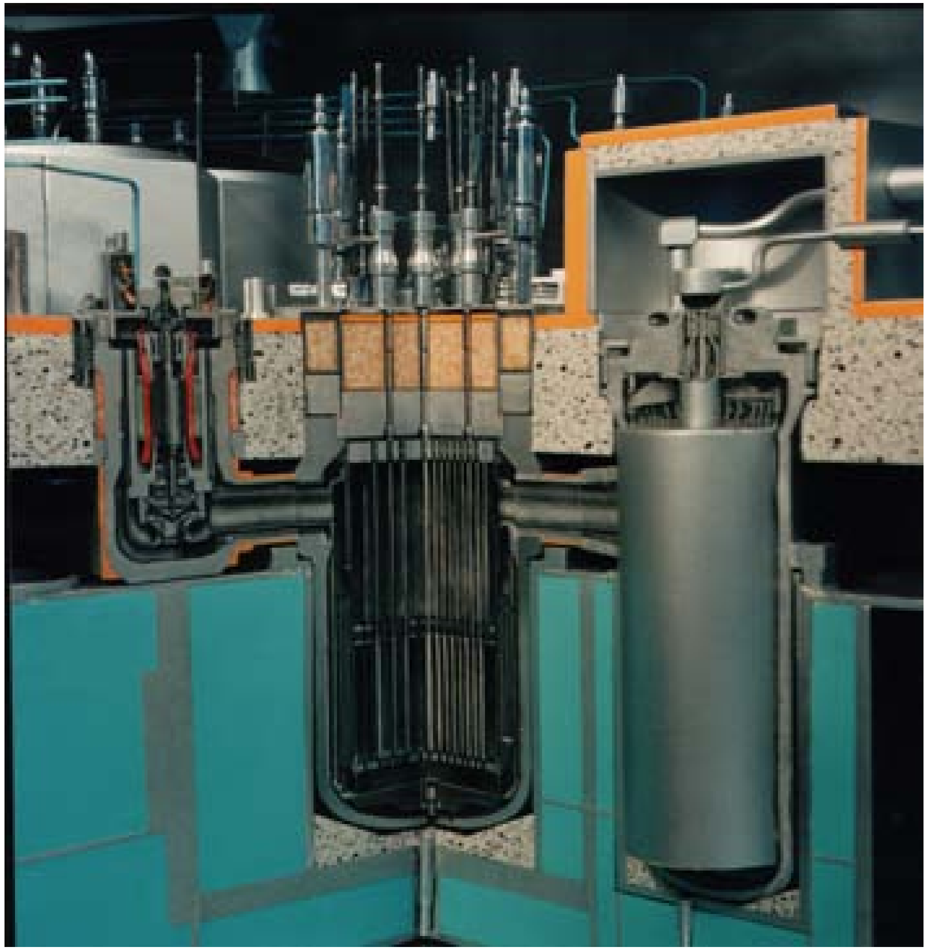


FIG. 6.6-3. General view of KLT-40S RP reactor room.

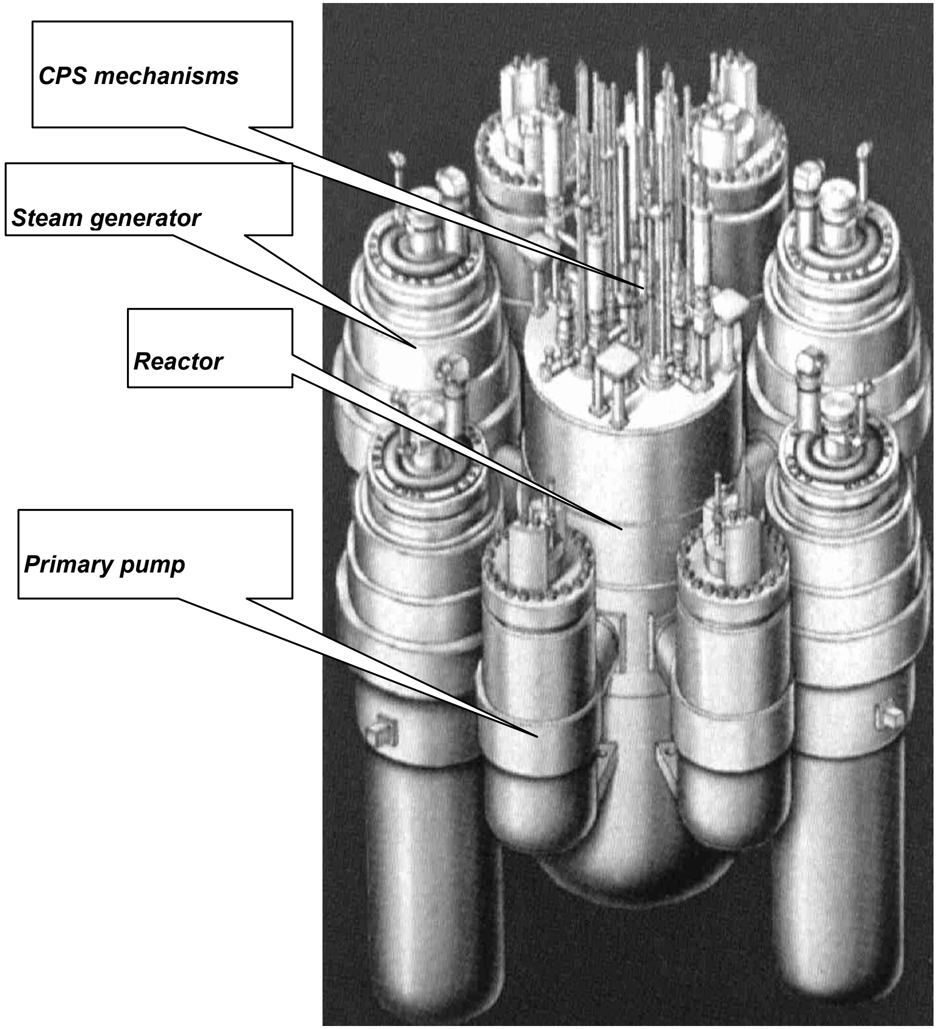


FIG. 6.6-4. Steam generating block.

The plant is based upon proven engineering solutions, as follow:

- PWR-type reactor being the most widely used in the world practice is applied;

- Reliable and proven elements are used in the reactor plant, along with elimination of “weak points” revealed by operation experience of analogs,

- Independent multi-channel systems are used for the reactor emergency shut down and passive and active heat removal.

6.6.2.1 Primary circuit system

The primary circuit system includes the main circulation circuit, the pressure compensation system and the purification and shut down cooling system.

6.6.2.1.1 Main circulation circuit

the main circulation circuit has a four-loop arrangement.

Reliable heat removal from the core is provided by:

- primary circuit circulation pump operation at large and small rate;
- operation of an electric pump for repair cooldown;
- natural circulation at cooldown.

To increase the safety level all primary circuit pipelines, including the pressurize, are connected to the reactor outlet chamber. This has a positive influence on the accident response for cases involvingloss of tightness of primary circuit pipelines.

6.6.2.1.2 Pressure compensation system

The pressure compensation system consists of four pressurizers and pipelines of high pressure gas system.

In the system of pressure compensation the following is provided in order to improve the life-time characteristics:

- screening of the pipelines profiled parts and welds, reactor vessel and pressurize nozzles;
- intermediate tank ipressurizepressurizer;
- spider-mixer, allowing to decrease thermocyclic stresses.

6.6.2.1.3 Purification and cooldown system

The purification and cooldown system consists of: 1st – 3rd circuits heat exchanger;
- 1st circuit filter;
- two electric pumps for repair cooldown;
- pipelines and valves.

6.6.2.2 Reactor core and fuel design

OKBM and VNIINM have developed the design of the core with a $3.3 \cdot 10^6$MW·hr power resource (22000 hours life cycle).

The core was developed on the basis of evolutionary decisions, on the basis of the fuel element in cylindric shell of zirconium alloy E-635 and intermetallic fuel with swelling compensator. Testing of such fuel element in the core of the "Yamal" nuclear ice breaker, as well as post-reactor testing in NIIAR have shown its high serviceability.

TABLE 6.6-II. MAIN CORE CHARACTERISTICS

Characteristic	Value
Life cycle, hr	22000
Power resource, 10^6 MW·hr	3.3
Duration of operation without refuelling (N=64-80%, τ=8000 hr a year), years	3.4-4.3
Fuel type	Uranium intermetallic in silumin matrix
Maximum depth of fuel burn-out, g/cm^3	1.29

A cassette structure for the core provides a proliferation resistant feature. It is possible to create such core with dioxide fuel, which is used in PWR. An alternative proliferation resistant feature is use of fuel with an inert matrix. Ceramic-metal fuel with uranium dioxide grains in aluminum or zirconium matrix, providing comparatively high specific uranium content (5 g/cm^3) is a prospective fuel. With this fuel, the power resource of the core will be 3.0-3.3·10^6 MW·hr, and the enrichment will not exceed 20%.

Main KLT-40S reactor plant equipment

The main equipment of KLT-40S reactor plant is unified with the equipment of KLT-40 plant.

Reactor

The reactor is a PWR with forced circulation in the primary circuit. The reactor consists of the vessel, head, removable block, core, Compensating Group (CG) drives and Emergency Protection (EP) actuators. The reactor vessel is forged and welded. The vessel and head of the reactor are made of thermally stable high-strong perlitic steel with anticorrosive facing.

6.6.2.3 Fuel handling and transfer systems

No information provided.

6.6.2.4 Primary components

Steam generator

A steam generator (SG) is a once-through heat exchanger of coil type with steam generation inside the tubes. Tube system of SG is made of titanium alloy in form of cylindrical coils. Steam generator vessel is made of low-alloyed steel with anticorrosive facing.

Primary circuit circulation pump (PCCP)

A leak-tight centrifugal one-stage pump with the shielded two-speed (two-winding) asynchronous motor is foreseen. The pump capacity is 870 m^3/hr at pump head of 0.38 MPa. Vessel structures of PCCP are fabricated of austenitic stainless steel and electric motor rotor is made of ferritic stainless steel. Lubrication and cooling of the bearings as well as electric motor rotor and stator cooling are provided by primary coolant which circulates by autonomous circuit, its heat is removed by cooling water.

Emergency protection actuator

Actuator of electro-mechanic type consists of rack mechanism with the spring, asynchronous electric motor and electromagnet. Drop time is not more than 0.5 s. Control members drop is provided by the spring in case of holding electro-magnet de-energization.

Compensating group drive

Compensating group (CG) drive consists of the screw-type mechanism of the reductor, step electric motor, displacement sensor, reference points sensor. At the electric motor deenergization CG screw descends under gravity from any position to mechanic stop. The CG drive elements are made of wear resistant stainless steel.

6.6.3 Description of the turbine generator plant systems

6.6.3.1 Turbine generator plant

The steam turbine plant (STP) is intended for the conversion of thermal power of steam generated in RP into electricity, and transport of thermal power for water heating in the intermediate circuit of the power and heat generation system. The FPU has two steam-turbine plants. Each steam turbine plant is autonomous and is connected to its RP. Main nominal characteristics, to which STP corresponds are presented in Table 6.6-III.

TABLE 6.6-III. MAIN NOMINAL CHARACTERISTICS

Characteristic, dimension	Value
Electric power at the generator terminals, MW	35
Thermal power transmitted to heat supply system, Gcal/hr	25
Steam parameters before turbo-generator set:	
- steam rate per one turbo-generator set at nominal mode, t/hr	220
- pressure, MPa	3.43
- temperature, ºC	285
Feed water parameters at RP inlet:	
- pressure, MPa	6.45
- temperature, ºC	170
- temperature of cooling water at the condenser inlet, ºC	10
Service life before the factory repair, years	10-12
Service life of main equipment, years	35-40

For RP arrangement and cooldown technological condensation plants are provided, they are located in the rooms of the reactor compartment. Heat supply to the coast is provided by intermediate circuit water circulating between FPU and the coast. The heating is provided by steam from the controlled turbine extraction. Intermediate circuit water is heated from 70°C to 130ºC in the pre-heaters, located in machine rooms.

Each steam-turbine plant includes the following main equipment and systems:

- turbo-generator set;
- main ejector and the ejector of suction system from the seals 1;
- two circulation two-speed electric pumps (each has 60% capacity);
- three feed electric pumps with the controlled electric drive (each has 60% capacity);
- one special feed electric pump;
- two circulation electric sweet water pumps of generator cooling system;

- deaerator;
- cross tank;
- four ion exchangers and two double mechanical filters;
- one condensate drain electric pump;
- regenerative heaters of feed water (low pressure heater No 1 and two vessel high pressure heaters No 2 and No 3);
- two main heaters of intermediate circuit water (each for 50% capacity);
- one peak water heater of intermediate circuit;
- air cooler of main ejectors flash steam, seal ejectors and deaerators;
- steam systems;
- condensate feed system;
- turbo-generator oil supply system;
- system of cooling by sea water;
- intermediate circuit system.
- following systems are common for two STPs:
  - system of receipt, pumping, delivery and separation of turbine oil;
  - system of receipt and pumping of feed water.

A turbo-generator set (TGS) is intended for the operation as a part of STP, which converts thermal power of steam, generated in RP into electric and thermal power for water heating in the system of power and heat generation. The TGS includes the following main equipment mounted at TGS frame:

- steam turbine with pressure reduction system, shaft rotating device, steam distributor, lock valves, control and protection system;
- electric generator;
- surface two section condenser with the expansion tank, condensate collector and safe diaphragm;
- three main condensate electric pumps (each for 60% capacity);
- two fresh water coolers of generators cooling system;
- pipelines and valves;
- level controller in the condenser;
- steam level controller in the seals;
- service site;
- TGS frame.

At normal operation modes the influence of STP upon RP depends on the change of TGS load. At the mode of joint STP and RP control depending on STP load, RP steam rating changes automatically by influence upon feed water flow rate. At the mode of separate control, RP operates with the fixed margin of steam rating, which is taken through the pressure reduction valve by the main condenser.

6.6.3.2 Condensate and feedwater systems

When forming steam condensate cycle, special attention was paid to the increase of its economical efficiency as for ship conditions. The system of regenerative feed water heating (in low pressure heaters, deaerator and high pressure heaters) was developed to provide 170°C temperature at RP steam generators inlet. Power take off for FPU needs is provided directly from main generators. For all main STP pumps (condensate, circulation and feed) electric drive was used. Condensate having high temperature is dumped directly to deaerator.

6.6.3.3 Auxiliary systems

Independent auxiliary systems are:

- two low pressure steam generators of 10 t/hr steam rating of saturated steam each at 0.5 MPa;
- two desalinating plants of 120 t/day capacity each;

- two technological condensation plants of 24 t/hr capacity each;
- one auxiliary boiler plant of 10 t/hr steam rating of saturated steam at 0.7 MPa pressure.

6.6.4 Instrumentation and control systems

Automated process control system

Control of FPU technological processes as well as control of FPU operation as an industrial object is provided by automated process control system (APCS) called "Laguna". The APCS of the floating power unit consists of the following elements:

- RP control and protection system;
- STP control systems;
- Electric energy system (EES) control systems;
- control systems common for all ship systems;
- radiation control systems;
- accidental events recorder;
- centralized power supply systems;
- fire alarm systems;
- ship born trainer;
- systems of industrial television;
- communication systems;
- administrative and economic activities systems.

Control of all types of FPU engineering means is provided from central control room (CCR) which has the following automated working places:

- of the shift head;
- of two RP and STP operators;
- of EES operator.

The possibility of RP transition into sub-critical state and control of cooldown systems at CCR removal from operation is provided from two posts of emergency cooldown, individual for each RP. Besides, the post-of-activities control (for the control of RP refuelling activities) and post-of-towage supervision (for the control of weather situation, fire safety and other ship parameters at FPU towage) are provided.

6.6.5 Electric (EES) system

The electric energy system (EES) of FPU consists of the system of electric energy generation and output to the coastal power system, house load power supply system and emergency power supply system.

System of electric energy generation and output to the coastal power system

The system of electric energy generation and output to the coastal power system includes:

- 2 main generators of alternating three-phase current of 35 MW nominal power each, of 10.5 kV stress, 50 Hz frequency together with the panels of excitation and control system;
- 2 main distribution devices of 10.5 kV nominal voltage;
- 8 transformers (4 of them are redundant) of 1600 kVA power and 15,5/0,4 kV voltage;
- 2 boards of electric energy output.

House load power supply system of normal operation

House load power supply system of normal operation includes:

- 4 redundant diesel-generators of 950 kV nominal power each;
- 4 main distribution boards of 400 V nominal voltage;
- 4 boards of low voltage of 230 V nominal voltage;
- 8 transformers of 100 kVA nominal power each and 400/230 V voltage.

All electric using equipment, located at FPU depending on its importance and influence upon safety provision belongs to one of four groups of reliability as for power supply.

The first group includes electric using equipment, which basing on safety condition does not allow the break in power supply for more than fractions of a second at all modes and which requires obligatory power supply after emergency protection (EP) actuation. Sets of constant power supply are used as such electric energy sources.

The second group includes the electric using equipment which allows power supply breaks for the time, determined by safety conditions and equal to some tens of seconds and which requires obligatory power supply after EP actuation. Such power sources are emergency diesel generators (EDG) with 10 s time of automatic start-up and loading.

The third group includes the alternating current equipment, which allows breaks in power supply at the disconnection of main electric energy power sources after EP and which provides safe modes of subsequent operation and necessary conditions of habitability. Such sources are reserve diesel generators (RDG) with 30 s time of automatic start-up and loading.

All the other electric using equipment belongs to the fourth group.

System of emergency power supply

For the distribution of electric energy to the electric using equipment of the groups I and II, an independent two-channel system of emergency power supply is provided for each reactor plant. The system of emergency power supply includes:

- 4 emergency diesel-generators of 300 kV nominal power each;
- 4 boards of emergency diesel-generators of 400/230 V nominal voltage;
- 4 boards of the reactor plant of 400 V nominal voltage;
- 4 sets of constant power supply of 2 kV nominal power;
- 8 transformers of 16 kVA nominal power each and 400/230 V voltage;
- 8 power supply boards of automatic system.

At the emergency situations, which cause main generators shut down, signals for RDG and EDG start-up are generated. After EP actuation turbo-generators operation at residual steam rating is provided. The RDG takes the load of electric using equipment without any break in power supply after synchronization mode realization. The EDGs are actuated and operate in the mode of idle running. At DG non-actuation power supply of the I-st and П-nd group electric using equipment is provided by the EDG.

6.6.6 Safety concept

6.6.6.1 Safety requirements and design philosophy

When designing KLT-40 S RP the following evaluated engineering decisions were used:

- Compact design of steam-generating block with short load-bearing nozzles between main equipment without large-diameter primary circuit pipelines;
- Use of developed reactor emergency shut down system actuators of various principles of operation:
  - fast acting EP rods;
  - compensating groups.
- Use of emergency heat removal systems connected to the first and second circuits;
- Removal of "week points" basing on the operation experience (improvement of primary circuit pressure compensation system, some steam generator units and the others);
- Use of the available experimental data, certified design codes and methods.

The following engineering measures are used for emergency reactor shut down:

- Two-set CPS;
- EP drop from self-actuating devices at the emergency primary circuit pressure increase;
- Use of liquid absorber system at the beyond-design accident.

Emergency core cooling is provided owing to:

- Location of primary circuit pipelines above the core with the connection to hot reactor chamber;
- Presence of passive cooling and protection system (hydraulic accumulator);
- Use of improved structure of recirculation channels.

Emergency heat removal from the reactor is provided by passive system of emergency cooldown in course of 24 hours at the loss of all power supply sources.

Containment system provides passive heat removal in course of 24 hours without medium discharge from the containment.

Main principles of safety provision

▪ Use of PWR with the developed inherent self-protection, which is provided owing to feedbacks in the core, thermal reactor sluggishness, natural coolant circulation and the others;

▪ Provision of functional and physical protection in depth due to effective safety barriers and localizing systems which exclude discharge of radioactive products outside FPU at the severest accidents accompanied by additional failures;

▪ Use of physically separated safety systems of active and passive principle of operation providing their putting into operation including by self-actuating devices without using external power sources and personnel intervention;

▪ Use of highly-reliable self-diagnosing automatic control systems and logic information systems of operator support;

▪ Use of diagnostic control systems, which allow to determine the real state and residual life of the most responsible equipment and pipelines of nuclear power plant;

▪ Provision of methods and means of severe accidents control.

6.6.6.2 Safety systems and features (active, passive and inherent)

The safety systems are arranged using reasonable combination of active and passive means with redundancy of necessary elements. The safety systems are:

- Emergency reactor shut down system.
- Emergency cooldown system.
- Emergency core cooling system.
- Emergency pressure decrease system.
- Containment.
- System of reactor cavity filling with water.

Emergency reactor shut down system

- Four independent groups of EP rods with their own drives actuated by passive principle from spring compression.
- Five independent compensating groups (CG) with their own drives, providing emergency reactor shut down after the control system signals:
  - drop with emergency rate from electric motor drive;
  - drop under gravity.
- Liquid absorber injection system (guard).

Emergency cooldown system

- Active cooldown channel through the heat exchanger of 1st circuit purification system with heat transfer to sea water through the 3rd circuit;
- Active cooldown channel through steam generators with heat removal to technological condenser and than to sea water with the redundancy of all active elements;
- Two passive cooldown channels through steam generators with heat removal to heat exchanger of emergency cooldown tank and than to the atmosphere by water evaporation from the tanks.

Actuation is provided both from the control system and by direct action of the medium pressure increase in the primary circuit with the help of hydraulically controlled air distributor. Water reserve in the tanks provides RP upkeep in safe state not less than 1 day without personnel intervention.

Emergency core cooling system

The system consists of two channels. Each channel meets the principle of single failure. The system includes subsystems of high and low pressure. High pressure subsystem includes passive (hydraulic accumulators) and active (water pumps and tanks) means of water supply to the reactor. The low pressure subsystem provides condensate return into the reactor with the help of recirculation pumps; this condensate is generated in the containment at the primary circuit loss of tightness.

Electric power supply breakers by pressure:

- Provide de-energization of CG drives electric motors:
  - by pressure increase in the primary circuit;
  - by pressure increase in the containment.
- Hydraulically controlled pneumatic distributor:
  - provides opening of valves with pneumatic drives of ECS passive channels by pressure increase in the primary circuit.

System of emergency pressure decrease in the containment

It is a system of passive principle of operation. Emergency pressure decrease is provided owing to steam condensation:

- In the bubbler tank;
- At the surface of heat exchangers, located in the upper part of the equipment room using water from emergency cooldown tanks;
- Surfaces of metal-water protection (MWP) tank and containment.

System of reactor cavity filling with water

The system is intended for reactor vessel protection against penetration in the severe beyond-design accident leading to the core dewatering and melting. Water and condensate is supplied to the reactor cavity from the bubbler tank, condensate accumulator and from the plating of MWP tank (owing to hydrostatic head). System operates by passive principle owing to steam condensation on heat exchangers of pressure decrease system and condensate gravity flow into reactor cavity. Besides condensate supply from the reactor room to the reactor cavity is provided by recirculation pumps.

Safety analysis

For SP NCGP on the basis of FPU with KLT-40S RP, the analysis of design and beyond-design accidents has been performed. Control of the accidents allows:

- To provide time margin enough for personnel actions on the accident control;
- To provide remote (from the board) or by place putting into operation of safety systems and additional means of accident control:
  - injection of liquid absorber;
  - use of stand-by feed pumps for water supply to steam-generator and into the reactor;
  - use of fire main for cooling water supply to heat exchangers of the third and fourth circuits and the others;
  - practically completely to exclude the possibility of “severe accident” with fuel damage owing to the presence of various and independent safety systems and means of accidents control.

Design accidents

Accidents with unpremeditated reactivity increase and heat removal failure run:

- Without allowable parameters deviations;
- Providing heat engineering reliability of the core;
- Without breaking of safe operation limits;
- Meeting safety criteria.

Accidents with loss of coolant:

- The core is under water level;
- Fuel elements overheating above nominal temperatures is absent;
- Containment provides coolant leakage from the reactor localization;
- Personnel intervention is not required in course of 8 hours from the moment of the initial event beginning;
- Critical doses of personnel irradiation are lower than allowable levels at which the performance of protective measures is required.

Beyond-design accidents

Preliminary situations with unpremeditated reactivity increase and heat removal failure run:

- Without the excess of maximum allowable primary pressure value;
- Without the excess of fuel elements cladding temperatures at which they lose their tightness.

At the accidents with primary coolant loss:

- The core remains under water level for the long time (not less than 1 hour);
- Radiation consequences are caused by the discharge of radioactive products, which are in primary coolant. There is no additional loss of tightness of fuel elements claddings.

Severe accidents (beyond design basis accidents)

Main strategy directions of severe accident control:

- Limitation of the core damage scope;
- Core melting prevention;
- Reactor vessel integrity upkeep with the retention of the core materials inside the vessel;
- Containment integrity upkeep taking into account the effects accompanying the severe accident;
- Limitation of radioactive products release into the environment.

Engineering means of FPU for accidents control in RP:

- Auxiliary systems (fire fighting, ventilation and the others);
- System of portable and washing water;
- Instrumentation including the system of radiation control;
- Non-standard means of power supply, water supply and air supply.

Engineering means for severe accident control

- Stand-by diesel generator;
- System of water supply to steam generators with low pressure steam dump from them for pressure decrease in primary circuit system and residual heat removal;
- Systems and means for water supply to the reactor.

Severe accident scenario

In accordance with the specifications requirements the problems of safety provision in postulated severe accidents are considered in the design of KLT-40S RP. Primary circuit system pipeline loss of integrity inside the containment with the failure of active systems of emergency core cooling is considered as an accident which may lead to overheating and melting of the core.

Considerable time margin and the possibility to perform correcting actions by personnel with the purpose to limit the consequences of the severe accident are provided by the following factors:

- The system of reactor cavity filling with water, which provides reactor vessel cooling to keep a core melt inside the vessel;
- Hydraulic accumulator and passive ECS:
  - provide slow accident development and create considerable time margins before the core melting;
  - perform the functions of pressure decrease system in the primary circuit and guarantee the exclusion of high pressure with core melting scenario.
- Enough margins of the containment design strength for long-term resistance to the loads (excessive pressure) taking into account hydrogen burning.
- The systems of emergency pressure decrease in the containment are used.

The characteristics of KLT-40S RP core, such as:

- Relatively low temperature of fuel melting (~1200-1400^{o}C);
- Not large volume of materials and power of heat release in the melt limit the value of accumulated energy and determine low thermal flows from the melt. The measures on the accident control and water supply to the reactor under these conditions will lead to the cooling and solidification of melted materials and stabilization of fuel state.

Radiation consequences of severe accident

Estimation of radiation consequences of the accident with severe core damage for the design value of the containment leakage, taking into account the localization of radioactivity release owing to the purification of the leak from the containment on the filters, shows that the zone of planned protective measures is limited by 500 m radius.

6.6.7 Plant layout

Supposed place of NCGP location

The place of SP NCGP is Severodvinsk town of Arhangelsk region. The planned place of SP NCGP on the basis of FPU with KLT-40 S RP is the East part of the area of water of Nikolskoe outfall at the economic moorage of FGUP “PO “Sevmash”.

Floating power unit (FPU)

The FPU is a smooth-deck non-self-propelled ship with the developed many-tier superstructure. In the middle part of FPU there is a reactor compartment and a compartment of nuclear fuel handling. Turbo-generator and electro-technical compartments are located to the bow direction from the reactor compartment and the compartment of auxiliary devices and living block are located to the stern direction of it. Such arrangement provides the required safety conditions and allows to make optimum lay-out of pipelines and electric cable. Table 6.6-IV gives the main engineering characteristics of FPU.

TABLE 6.6-IV. MAIN ENGINEERING CHARACTERISTICS OF FPU

Characteristic	Value
Main dimensions:	
- the largest length, m	144.2
- the largest width, m	30.0
- board height up to upper deck, m	10.0
- maximum draught in the midsection, m	5.6
Displacement with specification supply and removable cargo at coming to operation place after the end of construction, t	21500
Crew number	
- constant personnel	58
- temporary personnel	26
Docking periodicity, years	12
Specified service life of FPU, years	35-40

The FPU vessel has a shape of pontoon with the sharpened bow and straight stern. The pontoon has 3 decks, the second bottom and double board coming from the for peak to after-peak bulk-head and from the second bottom to the second deck by height. There is an elongated superstructure of 4-8 tiers on the pontoon. The FPU vessel is divided into 10 compartments by 9 water-proof bulk-heads, coming to the upper deck.

The FPU floodability is provided at the flooding of any two adjacent compartments for all specified cases of loading meeting the requirements of Russian Sea Navigation Register. The FPU is a determining component of NCGP, because it generates electric energy and heat and outputs it to the users through coastal infrastructure. Hydro-technical structures are intended for safe installation and fixing of FPU.

Coastal infrastructure consists of the structures and special devices intended for electric energy and heat receiving and transfer to the users at joint operation with FPU.

Storages of spent fuel assemblies, liquid and solid radioactive waste are located at FPU. The FPU has its own complex of reactors refuelling. All operations with radioactive substances are performed only at FPU. The FPU design provides five localizing barriers, excluding inadmissible discharges of radioactivity to the environment (see Fig. 6.6-5 and 6.6-6).

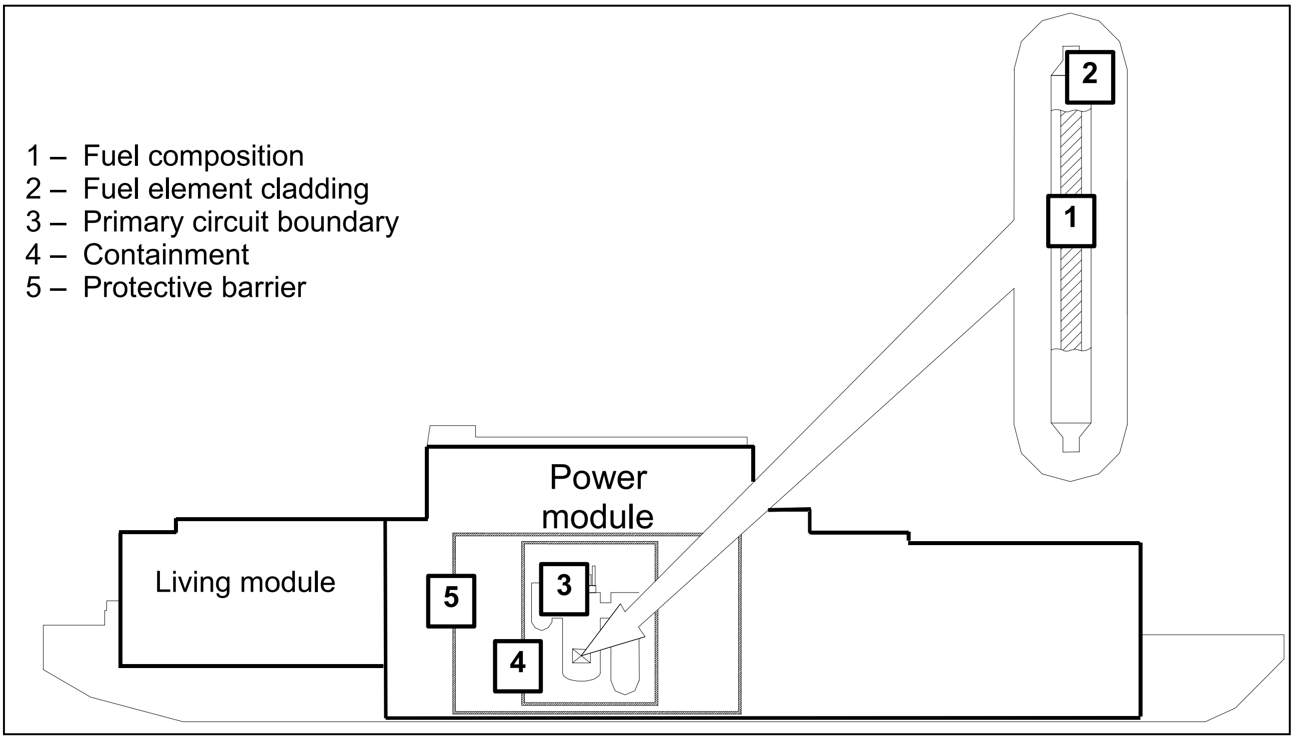


FIG. 6.6-5. Localizing FPU barriers.

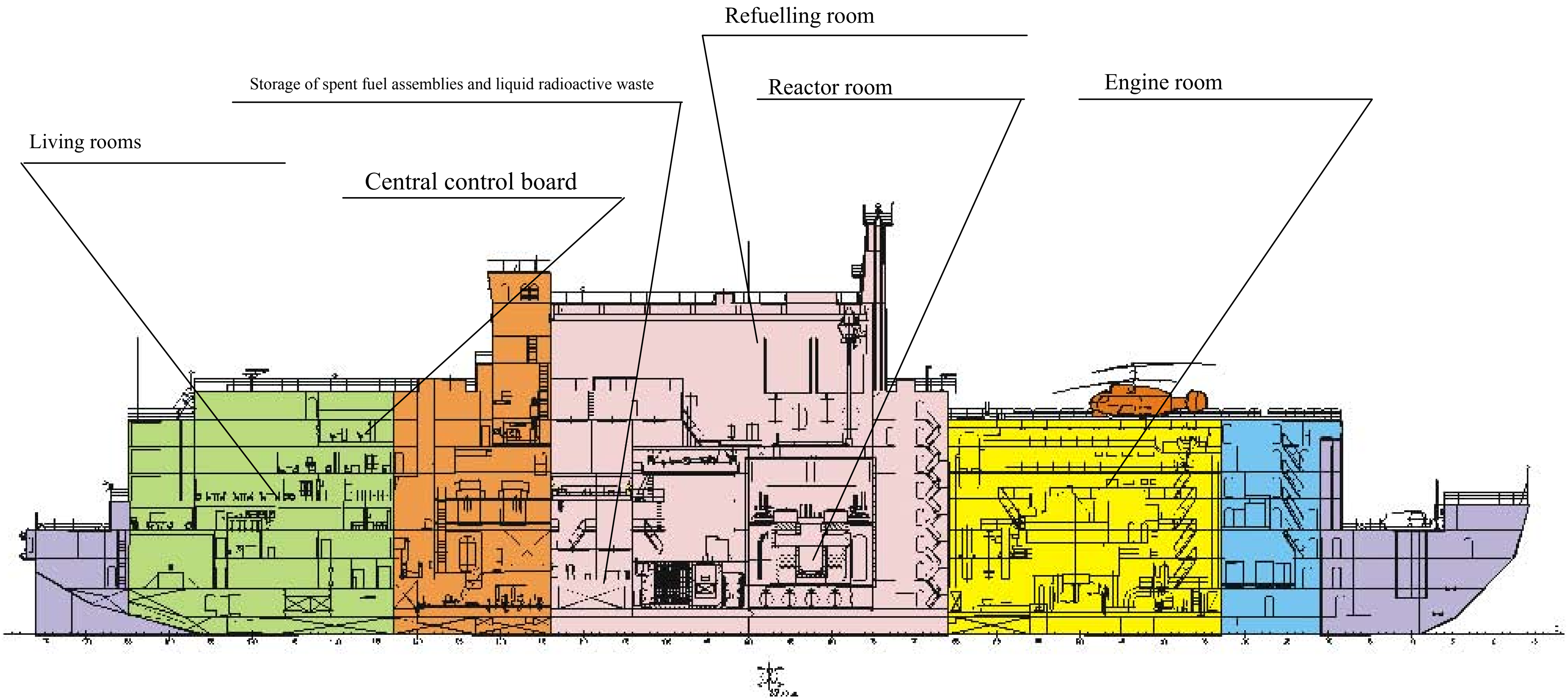


FIG. 6.6-6. Floating power unit.

Containment

A containment is a strong, tight compartment designed for 0.5 MPa (abs) internal pressure. There are no systems and equipment containing sea water inside the containment. At RP power operation, 300 Pa rarefaction is kept. To exclude containment break at emergency FPU flooding there is a special system, which provides containment filling and its subsequent pressurization. The containment is intended to perform the following main functions:

- Maintaining radioactive materials within the zone of accidents localization;
- Isolation from the environment of those systems and elements the failure of which may lead to inadmissible release of radioactive substances;
- Protection of personnel and population from ionizing radiation.

Containment is a localizing safety system. Containment elements include:

- Steel containment;
- Man-holes, locks, doors;
- Penetrations;
- Insulating devices;
- Relief and safeguard devices.

Each of two reactor plants of FPU is confined in a steel leak-tight containment that is a strong-tight vessel structure. The zone of accident localization confined by the containment is separated by its height with the help of leak-tight plating into two rooms – RP room and control room.

RP room contains the following main equipment:

- Reactor;
- Metal-water protection tank, which is a support structure for the reactor;
- Steam-generating units;
- Primary circuit circulation pumps;
- Pressurizers;
- Primary circuit filter;
- 1st-3rd circuits heat-exchanger;
- Drainage tanks of wasteless technology system;
- Hydraulic accumulators of emergency core cooling system;
- Bubbler tank.

Control room contains the following:

- Gas cylinders of pressure compensation system;
- Gas cylinders of high pressure gas system;
- Gas removal cylinder;
- Circulation pumps for gas removal;
- Condensate accumulators of reactor cavity filling with water system;
- Heat exchangers of emergency pressure decrease system;
- Stop valves of primary circuit system and the systems, hydraulically connected with the primary circuit.

Personnel access to the control room both at RP operation and during repair and refuelling is provided through the vestibule-lock, located at the level of the upper deck. Access to RP room is possible only at non-operating plant and is provided through two man-holes in the containment bottom, equipped with leak-tight covers (access through liquid radioactive waste room at the third deck level). Besides emergency exits (one from each containment) with leak-tight covers are provided in the longitudinal bulkheads of the containments, forming a cofferdam between them.

In the containment hold there is a man-hole for equipment transportation at RP refuelling and repair. At normal operation conditions the man-hole is pressurized and welded by special housing.

Pipeline communications and electric routes cross the leak-tight screen through special penetrations, providing leak-tightness of crossing places. Pipelines crossing the containment, through which radioactive materials may come outside the zone of the accident localization are equipped with isolating devices.

At all modes of normal operation, excluding short-term mode of sampling the sources of radioactivity and ionizing radiation in KLT-40S RP are within the limits of the containment.

The following loads were taken into account at the containment designing:

- External pressure at ship flooding; at the adopted design the external pressure withstood by the containment without loss of tightness is 0.94 MPa;
- Internal excessive pressure at the accidents, connected with the pipeline rupture in RP room or control room is 0.4 MPa. This pressure combines with temperature component: design temperature of steam-gas mixture is - 155°C in the control room and - 200ºC in the RP room itself.

Containment is divided by vertical-longitudinal cofferdam of 1.3 m width into two proof compartments symmetric relative to the center line.

Personnel access into the control room both at RP operation and during the refuelling is provided through the vestibule-lock. Besides, emergency exits (one from each containment) with leak-tight covers to the cofferdam located between them are provided in the longitudinal bulkheads of the containment. Access to RP rooms is possible only at non-operating RP and is provided through two man-holes, located in the containment bottom and provided with leak-tight covers.

Containment leak-tightness is taken in accordance with the requirements of "Nuclear Ships Classification and Construction Rules" of Russian Sea Navigation Register, and the allowable leak is not more than 1 % of containment volume per day. Given leak-tightness of the containment and peripheral biological shield provide radiation safety of personnel and population at the most radiologically dangerous design accidents and meeting of the requirements of standards and engineering codes.

Storage of spent fuel assemblies and solid radioactive waste (SFA and SRW)

Near the reactor compartment there is a compartment where the complex of nuclear fuel storage and handling systems is located; this complex consists of the following main elements:

- Storage of SFA and SRW;
- System of temporary location and handling of fresh nuclear fuel;
- Refuelling complex.

In SFA and SRW complex 2 stages of spent fuel storage are realized: wet storage in 3 autonomous leak-tight storage tanks, providing reliable removal of residual heat from SFA at the first moment after unloading from the reactor and subsequent dry storage in 4 autonomous tanks with air cooling. At FPU, there are two rooms for new fuel assemblies (NFA), each designed for 64 packing sets, providing loading of NFA of one RP. The rooms are intended for location of fresh nuclear fuel only for the period of its preparation for the loading to RP and do not provide storage at the other periods of FPU operation. A refuelling complex is used for performing activities on SFA unloading and NFA loading.

For liquid radioactive waste handling there is a complex of engineering means which provides accumulation of drainages from the systems servicing RP, radioactive water from biological shield rooms as well as temporary storage of LRW at FPU and its subsequent handling over to the support ships, or to the coast.

Special tanks located in the reactor compartment rooms and in the compartment for nuclear fuel handling allow to accumulate and store LRW generated during the whole overhaul period without its removal or processing at operation place.

6.6.8 Technical Data

No information provided.

6.6.9 Measures to enhance economy and maintainability

Usage of the reliable and safe series of the reactor plant KLT-40 as the prototype allows in many respects reducing the stages of RP design anddevelopment. Additional advanced highly effective safety systems, and reliable support systems, have allowed to create an independent source of power on the basis of reactor KLT-40, that meets the modern requirements for safety for NPP, fail-safe to equipment failures and personnel errors.

Long-term successful operation of marine reactor plants of similar type, and modernization and enhancement of RP equipment and systems provide improvement of the conditions of the plant in-service maintenance.

Basic economic indices of SP NCGP with RP KLT40S are given in the Table:

Performance	Value
Plant indices in the mode of electric power and heat co-generating: - released thermal power, Gcal/h (MW) - released electrical power, MW	 2×25 (29) 2×30
Maximum coefficient of average plant loading per year	0,85
Plant service life, years	40
Capital outlay for construction 10^6 $	165
Specific capital outlay, $/kW$_e$	2360
Cost price of 1 kW·h of released electric power, cent/kW·h	3,629
Cost price of 1 Gcal of heat, released to the consumer, $/Gcal	9,072
The boundary rate: - for the electric power, cent/kW·h - for heat, $/Gcal	 4,0 10,0
Fuel constituent in the cost price of electric power, cent /kW·h	0,898

The SP NCGPs are universal by the possibilities of their application, ensuring electric power generation parallel with a possibility of heat supply, for sea water desalination included.

The term of delivery for the pilot SP NCGP with RP KLT-40S on a turnkey basis will not be more than 5 years from the moment the Contract has been signed. Within 2,5÷3 years manufacture of commercial plants can be set up for delivering 2÷3 plants every year, the quantity determined by the needs of the Customers.

Capital outlay for construction of SP NCGP with RP KLT-40S amounts to 165 x 10^6. $ (by the end of 2002, the rate of exchange being 31,6 rbl./1$).

The cost of construction of SP NCGP with RP KLT-40S is broken down per years in accordance with the general schedule of construction.

The schedule of financing the construction of SP NCGP with RP KLT-40S:

Years of construction	1	2	3	4	Total
Capital investment, 10^6\$ - NCGP with RP KLT-40S	34,5	52,2	59,1	19,2	**165**
Capital investment, % - NCGP with RP KLT-40S	20,91	31,64	35,82	11,64	**100**

Breakdown of expenses for construction of SP NCGP with RP KLT-40S is made up of the following constituent items:

Clause	KTS-40S expenses	
	10^6\$	%
Equipment	80,64	48,87
Construction and erection activities	69,04	41,84
Others	15,18	9,19
Total	**165**	**100**

The enlarged structure of operational expenses for the floating NCGP with KLT-40S reactor plant consists of the following expense items:

Average annual expenses for the service life of NCGP	Sum, thousand \$	%
Depreciation charges	4125,0	25,22
Expenses for fuel with delivery and overload + spent fuel management + radwastes	4098,535	25,06
Wage fund	654,948	4,00
Overhead expenses (maintenance work + overhaul)	5830,142	35,65
Taxes (at boundary electricity rate 4,008 cent/kW·h)	1645,501	10,06
Complete annual costs (without purchasing first load core)	**16354,126**	**100**

6.6.10 Project status

- In accordance with the inquiry of FGUP “PO Sevmash” to Russian Minatom (of 08.12.1999 No 89.01.01/3031), in August 2000 an “Agreement on Intentions” was signed between Russia Minatom, Concern “Rosenergoatom”, Russian Agency on Shipbuilding and GUP “PO Sevmash”. This agreement foresees the joint actions on the realization of the construction Project of floating power units with reactor plants KLT-40S for small power nuclear co-generation plants and nuclear power desalination complexes. Respectively, a “Declaration on Intentions” for small power NCGP construction on FPU basis at the site of FGUP “PO Sevmash” was developed and agreed.

- Russia Minatom signed an order “On the realization of small power lead nuclear co-generation plant construction project on the basis of floating power unit with KLT-40S reactor plants in Severodvinsk of Arkhangelsk region” (of 30.11.2000 No 739). The Federal Board of medical, biological and extremal problems at Russian Ministry of Public Health gave “Radiation and hygienic substantiation of radiation safety category of FNCGP located at FGUP “PO Sevmash” of 24.12.2001.

- Severodvinsk Municipal Council approved the decision “On the location and construction of small power nuclear co-generation plants on the basis of floating power unit with KLT-40S reactor plants in Severodvinsk” of 21.03.2002 No 28. The Archangelsk regional group of Russian political and public movement “Russian Greenpeace Movement” (Severodvinsk town department) signed “Resolution of public ecological examination of the construction project of SP NCGP on the basic of floating power unit with KLT-40S reactor plant in Severodvinsk of Archangelsk Region” of 15.04.2002 No 04-03/13. The Arkhangelsk Region Assembly of Deputies issued a Resolution “On the support of construction and location of small power nuclear co-generation plant on the basis of floating power unit with KLT-40S reactor plants in Severodvinsk" of 28.05.2002 No 279.

- State sanitary and epidemiological inspectors of Russia (Federal Board of medical, biological and extremal problems at Russian Ministry of Health) signed a Resolution for the final design 20870 with KLT-40S reactor plant "Small power nuclear co-generation plant on the basic of floating power unit" of May 24, 2002 No 21-02.

- Russia Gosatomnadzor signed "A Decision on the subsequent consideration of documents for obtaining Russia Gosatomnadzor License for location in Severodvinsk of Arkhangelsk region of small power nuclear co-generation plant on the basis of floating power unit with KLT-40S reactor plants" of 17.06.2002 No 6-38/609.

- Russia Gosatomnadzor (NTTS YARB) signed "Expert resolution on safety of FPU with KLT-40S construction by FGUP "PO Sevmash" of 11.06.2002 DNP-5-363-2002.

- RF State Committee of fishing (Central Board on industrial fishing examination and norms of fish reserves protection, reproduction and accumulation) signed "Resolution on the materials of feasibility study of SP NCGP location and construction in Severodvinsk of Arkhangelsk region" of 21.06.2002 No 03-01/309.

- Department of NPP physical protection of the Concern "Rosenergoatom" signed "Opinion on the solutions on physical protection system of NCGP on the basis of floating power unit in Severodvinsk" of 31.05.2002 No 11-11/203.

- Ministry of RF Natural Resources (State Service of the Environment Protection "Rosecologia" signed an Order "On the approval of expert committee of state ecological examination resolution "Feasibility study of SP NCGP construction on the basis of floating power unit with KLT-40S reactor plant in Severodvinsk of Arkhangelsk region"" of 18.07.2002 No 447.

- State Concern "Rosenergoatom" sent documentation for SP NCGP location licensing to RF GAN.

Conclusion

The NCGP on the basis of FPU with KLT-40S RP which is an up-dated design of a nuclear plant for an ice breaker may be constructed at low expenses and in a short time. It is a safe and acceptable electric power and heat source from the point of view of technical and economical indices for the North regions and some regions of the same status even if they have their own power sources.

A minimum influence upon the ecological system at the construction and ecological compatibility at operation as well as high competitiveness as for technical and economical indices with the alternative power sources of comparative power are important arguments for the benefit of NCGP realization on the basis of FPU in various Russia regions.

The FPU with KLT-40S RP is a near term option for the generation of electric energy and desalination of seawater in different world regions.

6.7 PSRD-100 (JAERI, JAPAN)

6.7.1 Introduction

From the view point of addressing global warming and energy security, it is necessary to increase the utilization of nuclear energy for not only electricity generation by large scale nuclear power plants, but also usages such as heat supply to air conditioning, sea water desalination etc. A "Long-term Program for Research, Development and Utilization of Nuclear Energy in Japan" established in fiscal year 2000 by the Japanese Atomic Energy Commission proposes that more innovative reactor technologies including reactors for heat supply should be developed.

The U.S. Department of Energy, DOE, surveyed the feasibility of deployment of small, modular nuclear reactors in remote communities, such as in Hawaii and Alaska, that are deficient in transmission and distribution infrastructures (DOE, 2001). The thermal outputs of reactor plants required fell in a range of 60 to 200 MWt for most of the cases. Some innovative small reactors having enhanced safety and competitive economics were evaluated to be feasible for deployment in a decade. These small reactors basically would also be feasible for Japan and other countries.

The Japan Atomic Energy Research Institute (JAERI) completed the basic design of an advanced marine reactor for large ships, MRX (Kusunoki, et al., 2000), which was evaluated in the report of the DOE as the one of feasible reactors for deployment in remote areas.

On the base of MRX design, the JAERI has been developing a concept of PSRD (**P**assively Safe **S**mall **R**eactor for **D**istributed Energy Supply System) that is used as energy supply source for various utilizations such as small grid electricity generation, heat supply, and seawater desalination. The reactor thermal power of the PSRD is 100 to 300 MWt, taking account of deployment not only in small islands in Japan, but also in large ones including Okinawa.

High priority in the design is laid on further enhancement of safety as well as improvement of economy, since the reactor should be sited close to energy demand areas and with various types of siting. To comply with them, the possibility of accident occurrence is extremely reduced and the fully passive safety systems are adopted. System simplification contributes to enhancement of safety, but also improvement of economy through reducing costs of construction, operation and maintenance. Lōng-term operation without refueling or maintenance is an important factor for economy improvement. The reactor core is designed to be capable of operating continuously for five years with a full power. A study on economy of the modular reactor systems with PSRD generating electricity is now continuing. In the design, innovative technologies are adopted, some of which have already been developed and others that are under development.

6.7.2 Description of the nuclear systems

6.7.2.1 Primary circuit and its main characteristics

A cross section of the reactor pressure vessel (RPV) together with that of the containment vessel (CV) is shown in Fig. 6.7-1. Inside the RPV, the core is located in the lower part, the steam generators (SGs, two sets) in the middle part, and the in-vessel control rod drive mechanisms (INV-CRDMs) in the upper part. Around the core outside the core barrel, a radiation shield is provided. There are neither the primary coolant pumps nor the pressurizer.

The SG is of the once-through, helical coil tube type. The primary cooling water flows outside the tubes, and the secondary water and the steam flow inside the tubes. The SGs are hung from the main flange of the RPV. In refueling, the center flange together with the INV-CRDMs after de-latching the control rods is removed. The primary cooling water flows up after passing through the core by a single-phase natural circulation driving force, turns out the core barrel through the flow holes, which are positioned above the SGs, and flows down through the SGs. The water level will vary during normal operation between the top and the bottom of the flow holes.

The volume control system and the purification system are not used during reactor power operation, in order to simplify the system and reduce possibility of a loss of coolant accident due to pipe rupture. These systems, however, will be used except for the reactor power operation, e.g., prior to opening of the RPV cover for refueling or prior to reactor startup after closing it. These lines of the system will be completely isolated during the reactor power operation. Pipes penetrating the RPV are limited to only the pipes of the steam, the feed water and the safety valves.

The CV is filled with water, i.e., water-filled containment. There is the nitrogen gas in the upper space and the water below the gas inside the containment. The RPV, the emergency decay heat removal system (EDRS) and a containment water-cooling system (CWCS) are submerged in the water. The water-filled CV has a function of safety engineered system as well as one of enclosing the area for prevention of radioactive material release to the surrounding. The water inside the CV has also the role of radiation shielding instead of the concrete shield.

A thermal insulation is necessary to prevent the heat loss from the RPV into the water. The RPV is covered with a water-tight shell (WTS) made of stainless steel. Between the RPV and the WTS, thermal insulation of the stainless steel felt is inserted. The heat loss from the RPV with this insulation is estimated less than 1% of the rated power.

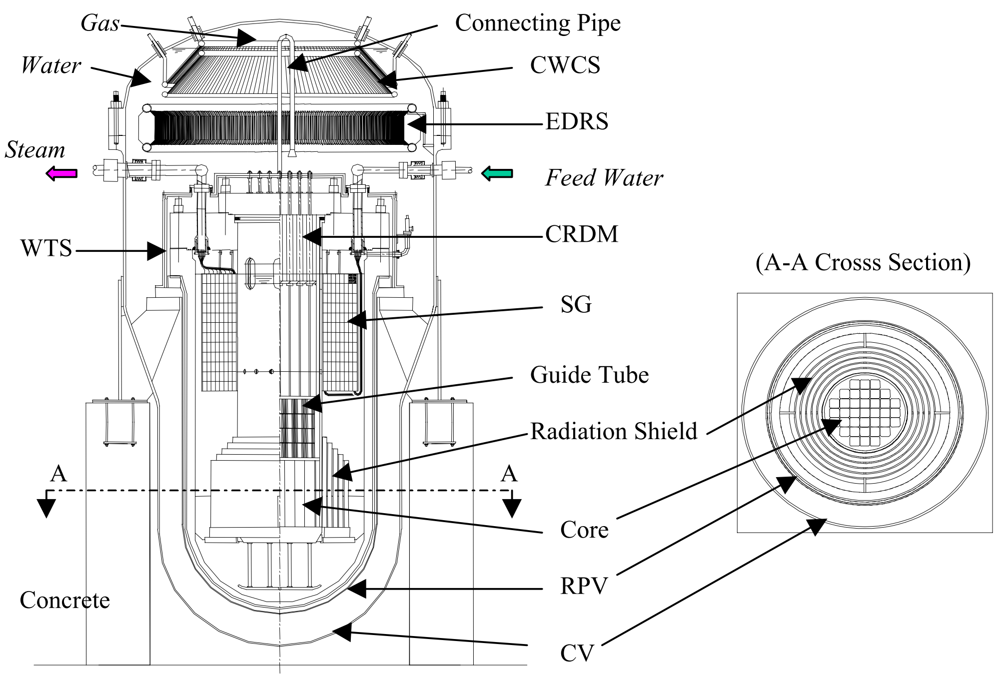


FIG. 6.7-1 Concept of PSRD

6.7.2.2 Reactor core and fuel design

The core of PSRD has been designed so as to achieve long life operation over five years without refueling or shuffling for enhancement of economic competitiveness.

Design conditions were set up as follows:- i) Low enriched UO_2 of which ^{235}U enrichment is to be less than 5% on the base of a framework in the current regulations of Japan;- ii) The chemical shim for the power control in the normal operation and the boron injection for reactor shutdown in an emergency are not adopted in order to simplify the system and reduce the pipes penetrating the RPV wall;- iii) The excessive reactivity at EOL is to be larger than 2% for immediately re-startup operation;- and iv) The reactivity shutdown margin at condition of a cold state is to be larger than 1%.

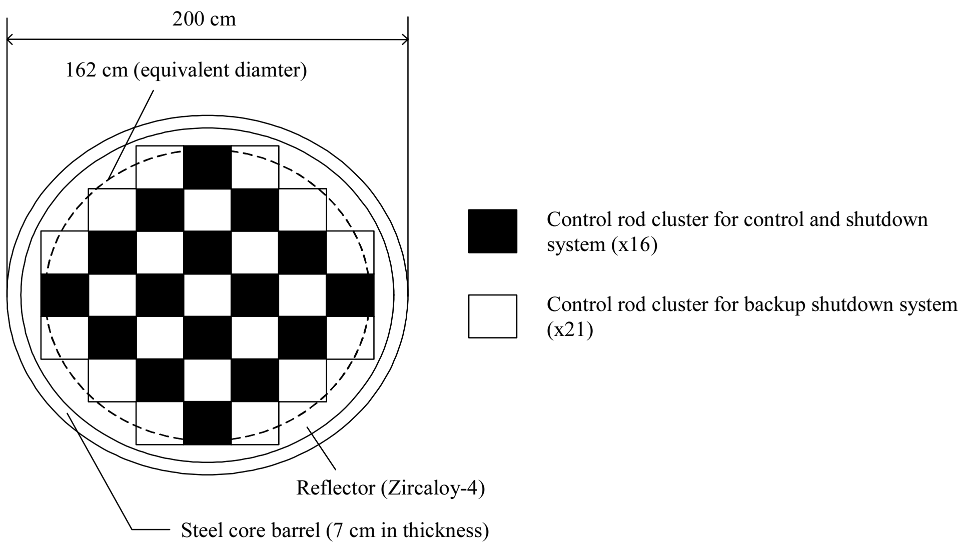


FIG. 6.7-2 Cross sectional view of PSRD-100 core

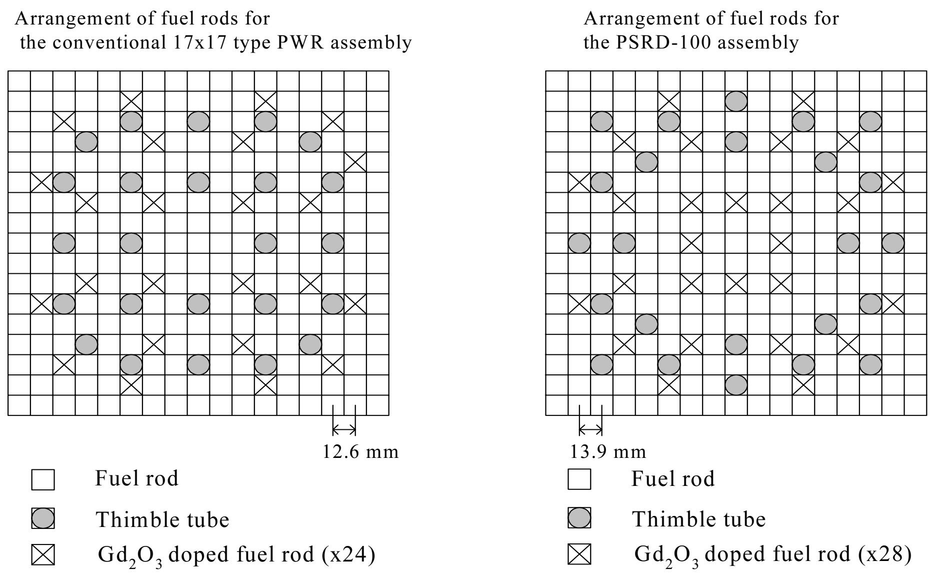


FIG.6.7-3 Arrangement of fuel rods in assembly

A cross sectional view of the PSRD-100 core is shown in Fig.6.7-2. The equivalent diameter is 162 cm and effective core height is 150 cm. Dimension of the core was determined from the thermal output, average linear power density, and pin pitch so that the reactor core can be installed inside the core barrel whose outer diameter is 200 cm. Space between the core barrel and outside the fuel assembly is filled with zirlaoy-4 reflectors. The reactor core consists of 37 fuel assemblies and control rods as the reactor shutdown system would be installed to all assemblies to maintain enough shutdown margin. The reactor shutdown system is divided into two systems: reactivity control and shutdown system with 16 control rod clusters, and backup shutdown system with 21 control rod clusters. All control rods for backup shutdown system are withdrawn during normal operation.

In the present design, the fuel assembly is based on that of current PWRs, of 17×17 type of fuel assembly with Zircaloy-4 cladding UO_2 pellets. Fuel pin pitch (13.9mm) of the PSRD core, however, is wider than that (e.g. 12.6mm) of the current PWRs to ensure efficient burn-up by greater moderation (Fig. 6.7-3).

An issue concerning the high burnup of core is how to suppress a rather large reactivity at the BOL. To comply with this, the PSRD adopts the fuel rods doped with Gd_2O_3 as well as the control rods that can be inserted in all fuel assemblies.

The nuclear characteristics were evaluated by core analyses with SRAC95 developed by JAERI (Okumura, et al., 1996), which contains the ASMBURN module for assembly calculation and the COREBN module for core burnup calculation.

Burn-up characteristics of the core are shown in Fig. 6.7-4 for a condition of continuous full power operation. The core operating cycle is 5 years with the core load factor of 100%.

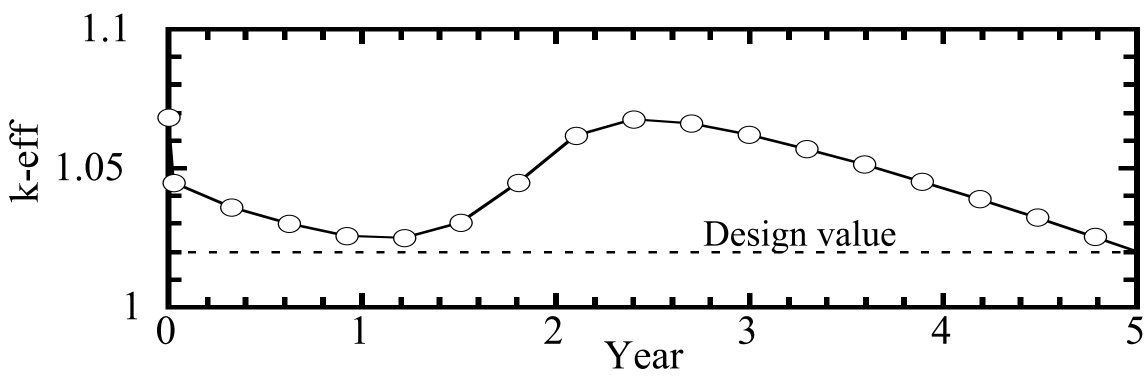


FIG. 6.7-4 Burn-up of PSRD 100MWt

6.7.2.3 Fuel handling and transfer systems

An exclusive barge is used for refueling or periodical maintenance of reactors. After refueling, the spent fuels will be stored in the pit, beside the reactor containment. If necessary, the spent fuels will be transported to a factory for retreating.

6.7.2.4 Primary components

The reactor pressure vessel accommodates the reactor core, the steam generators and the reactor pressure vessel internals. The design pressure of the pressure vessel is 11MPa, and the diameter and the height of it are 4m, and 10m, respectively. As described above, there are not the circulating pumps and no pressurizer in the PSRD.

The steam generator with two sets is the type of one-through helical tube of inner diameter 14.6mm and outer diameter 19mm. The total number of heat transfer tubes is 465. The primary loop fluid flows outside the tubes, and the secondary fluid flows inside the tubes, taking account of flow resistance of the natural circulation in the primary loop.

The control rod drive mechanisms are installed in the upper space inside the reactor vessel (INV-CRDMs). A control rod ejection accident, therefore, does not need to be considered as one of the design basis accidents for this reactor.

6.7.2.5 Reactor auxiliary systems

The chemical and volume control systems are not used during reactor power operation in the PSRD in order to simplify the system and reduce possibility of a loss of coolant accident due to pipe rupture.

The amount of radioactive corrosion products during plant operation and the dose rate due to it is evaluated to be small. Water volume change of the primary loop between a cold shut down state and the normal operation state can be absorbed in a relatively large free volume inside the reactor vessel even if without the volume control system.

6.7.2.6 Operating characteristics

The PSRD is designed for being automatically operated between 10 to 100% of the rated power. The plant control scheme is based on the “reactor follows plant load”. The reactor power control is done by only the control rods, but not with chemical shim.

In start-up of the reactor from the cold state, a nuclear heating-up will be taken because there are neither the circulating pump nor the pressurizer for heating.

6.7.3 Description of the turbine generator plant systems

The main steam system of the PSRD for generating electricity is similar to that of currently operating LWRs, and the turbine generator plant consist of a high pressure turbine and a low pressure turbine.

The operation pressure and temperature (10MPa, 584K) in the primary loop of the PSRD are lower than those of existing large size PWRs. The reasons are that even if these operation conditions are upgraded to those of the existing PWRs, the increase of the thermal efficiency is very small, but the thickness of the RPV and CV should be increased, resulting in high construction cost.

Electricity output depends on requirement of utilities, i.e., whether they want a constant full power to be supplied or to respond to load change. While for the former requirement, adoption of the regenerative cycle of steam turbine can produce electricity by the thermal efficiency of about 31%, for the latter, the system can achieve only about 27% because the regenerative cycle is not practical due to rather complicated handling.

The JAERI has studied a feasibility of deployment of the PSRD in Okinawa, where the electricity is generated by the boiler or the gas turbine using the fossil fuels and their capacity is 1,825MWe in all in 2002. The base load of the electricity is 365MWe, equivalent to 20% of the capacity. To generate 365MWe, 12 modular units of PSRD with each 100MWt can supply the collecting steam from each the steam generators to one turbine. Since efficiencies of the turbine and the generator increase with their capacities, this system is advantageous to that of small capacity turbine for each steam generator.

6.7.4 Instrumentation and control systems

6.7.4.1 Design concept, including control room

The control rod system has two groups of clusters, namely, buck-up shutdown and power control groups. Both of them are driven by in-vessel type control rod drive mechanisms (INV-CRDMs) that are installed inside the reactor vessel. The JAERI has developed the INV-CRDM driven by electric motor for the MRX and PSRD (Ishida et al., 2001) shown in Fig. 6.7-5. The INV-CRDM works under conditions of the high temperature water in the MRX and the steam in the PSRD. The control rods of power control group are moved automatically to keep the core outlet temperature constant, that is, the constant pressure.

The signals for reactor control and protection are the reactor power, the temperature of core outlet, the pressure of containment, the water level of containment, the temperature of containment, the radioactivity dose rate, etc.

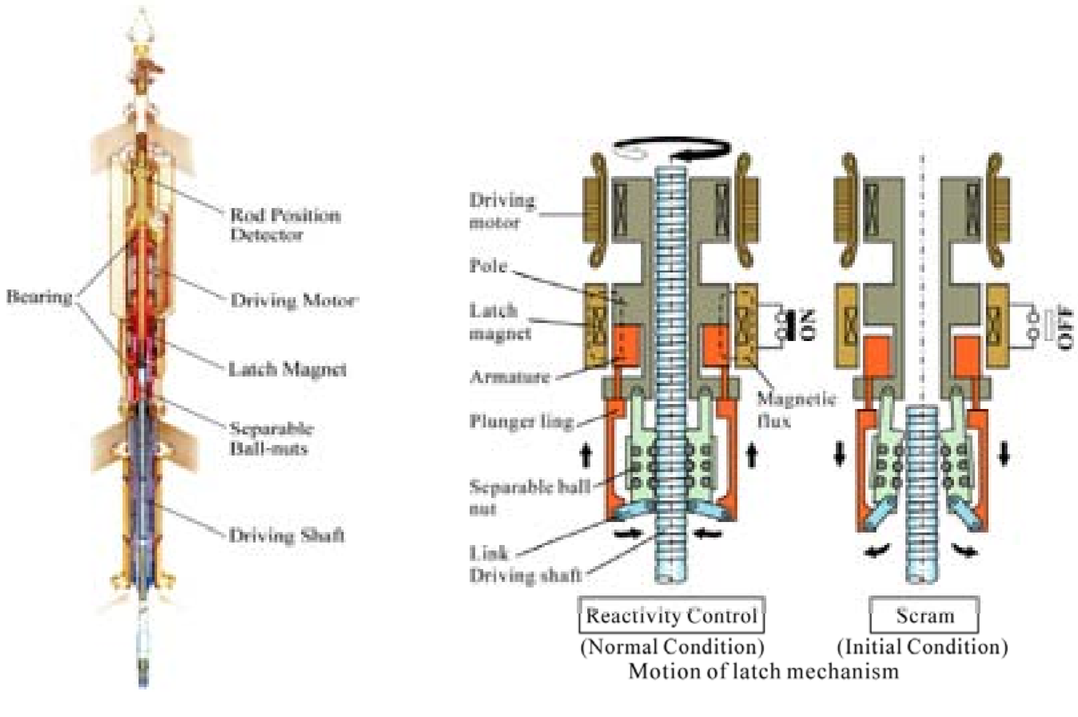


FIG.6.7-5 Concept of INV-CRDM

6.7.4.2 Reactor protection and other safety systems

In case of unusual conditions, the reactor protection system will work and the control rod will insert into the core for scram. The PSRD employs a passive reactor shutdown system. This passive shutdown system is an additional back-up system.

De-latching the driving shaft connecting the control rod by the INV-CRDM can scram the reactor after receiving a scram signal: This is the active reactor shutdown. The reactor can be also shutdown passively by inserting the control rod into the core as follows. Two types of concepts of passive reactor shutdown devices, which are attached at the driving shaft of the INV-CRDM. The one is for response to core temperature rise, PASD-HT -Passive Actuation Scram Device for High Temperature-. The device consists of a permanent magnet, a magnet enclosure and a magnet control plate. In normal operation, the control rod cluster is adsorbed to the driving shaft by magnet force at the contact surface (or the separation surface).

The magnet control plate is made from magnetic substance that has a characteristic of the saturated magnetic flux depending on the temperature; its flux decreases very much at high temperature e.g., about 620K. When the core outlet flow of very high temperature passes through the device, the magnet contacting force decreases due to drop in the saturated magnetic flux of the magnet control plate and the control rod cluster separates by gravity force from the driving shaft.

This concept is similar to that of SASS -Self Actuation Scram System- applied to some fast breeder reactors. The difference between the two, is that while the SASS uses an electric magnet coil with the moveable cable, the PASD-HT does the permanent magnet and allows the system to be simpler.

The other is for response to water level drop inside the RPV, PASD-LW (Passive Actuation Scram Device for Low Water level). It consists of a magnet coil, a magnet enclosure and the magnet control plate, the same as the one of PASD-HT. In normal operation, this device is surrounded by water and the control rod shaft is contacted to the driving shaft by magnet force produced by an electro-magnetic flux. When the water level drops and the device is filled with steam, the temperature of the magnet control plate rises owing to decrease of heat removal to the steam, and the contacting force decreases due to drop in the saturated magnetic flux of the magnet control plate. In the same way, the control rod shaft separates from the driving shaft.

These passive reactor shutdown devices are also available to the other type of CRDM such as the ex-vessel CRDMs of the current LWRs.

6.7.5 Electrical systems

6.7.5.1 Operational power supply systems

No information provided

6.7.5.2 Safety-related systems

No information provided.

6.7.6 Safety concept

6.7.6.1 Safety requirements and design philosophy

Since the PSRD is set close to energy demand areas and with various types of siting, radioactive material should not be released to the surroundings even in the event of hypothetical accidents including earthquakes. The safety concept of the PSRD is, therefore, based on prevention, as possible and mitigation of accidents.

The pipes penetrating the reactor vessel are limited to only those of the steam, the feed water and the safety valve, and consequently the frequency of LOCA occurrence is very low. In case of an accident, mitigation of it will be done automatically by passive safety systems.

6.7.6.2 Safety systems and features (active, passive, and inherent)

The PSRD employs passive reactor shut down systems and an engineered safety system. The engineered safety system, the passive safety system, consists of the water-filled CV, the EDRS with hydraulic force valves and the SGs, and the CWCS, as shown in Fig. 6.7-6. The main functions of this system are to maintain core flooding in cases of accidents including a LOCA and to remove the core decay heat.

Core flooding can be maintained passively by pressure balance of the containment and RPV in an early transient period of a LOCA, and with the help of EDRS and CWCS in a later transient period. Thus, the core flooding in a LOCA can be attained passively without ECC pumps or an accumulator.

Decay heat can be removed with the EDRS as follows. When an accident, e.g. a LOCA happens, the reactor is shut down, the feed water pumps stop and the isolation valves for steam and feed water lines close. Hydraulic force valves, force of which are supplied through pipes from the feed water pump outlets, open passively due to pump stop, to flow the cooling water of the EDRS to the SGs. The core decay heat is transferred to water of the CV through the SG and EDRS, by natural circulation heat transfer mode. Heat is transferred from water of the CV to the atmosphere, through the heat pipes of CWCS.

The pressure increase of the containment in case of a LOCA can be suppressed by the steam condensation phenomenon in the water-filled containment. The design pressure of the containment, therefore, can be greatly lowered. The pressure increase and the water level of the core in a case of LOCA depend on the initial water level and the gas volume of the containment.

The relationships between the initial water level of the containment and the balance pressure, etc., were studied experimentally (Kusunoki, et al., 1998). In the basic design of the MRX with the thermal power of 100MW, the above-mentioned performance of the water filled CV was analytically confirmed.

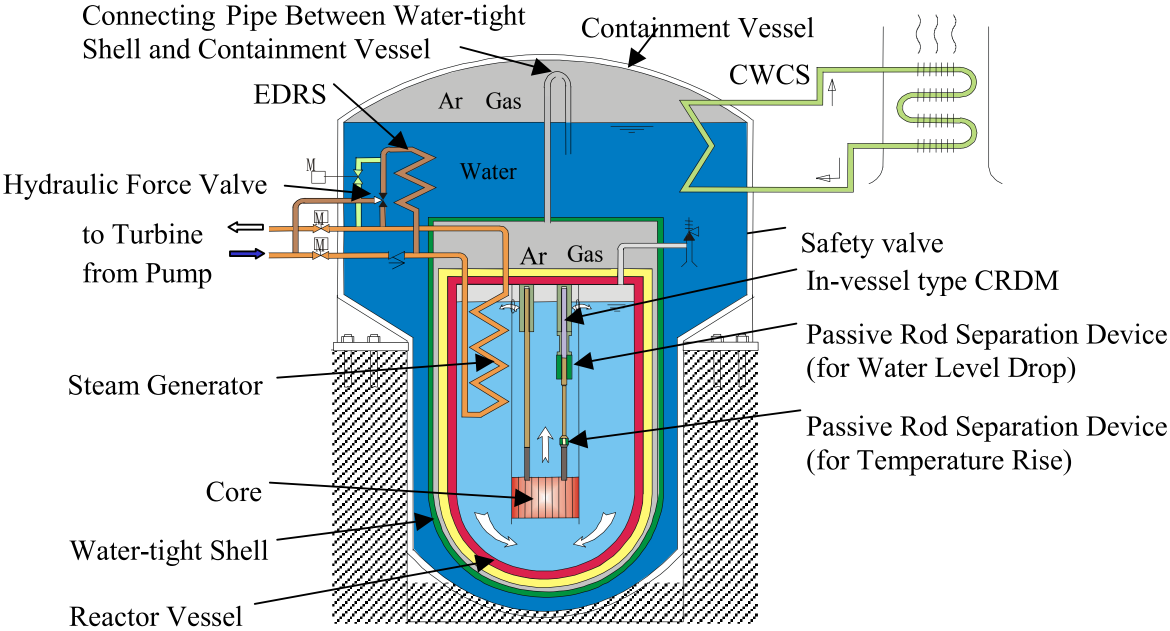


FIG.6.7-6 Passive safety system of PSRD

The water-tight shell (WTS) has a U-bend pipe connecting to the water inside the CV. In case of a LOCA due to pipe rupture inside the WTS, steam or gas will be discharged through the pipe into water and steam will condense there to suppress pressure rise of the CV. The pipe has a very small hole at the top of the U-bend pipe, which allows to flow between the space inside the WTS and the upper gas space of the CV according to temperature change due to the states of reactor operations such as the full power, or the cold shutdown.

It follows that the engineered safety system of PSRD is a fully passive and very simple system.

6.7.6.3 Severe accidents (beyond design basis accidents)

If a severe accident - a core damage accident - happens, cooling of RPV will be effective even from outer-side with filled water: RPV outer-cooling type of In-vessel Retention (IVR). Although the water is always provided inside the containment, the water is separated by the WTS from the RPV. If a hole is made at the bottom of the WTS in case of a severe accident, the water will enter through the hole with the help of the pipe for breathing as shown in Fig. 6.7-6. The JAERI is currently performing a study for a simple device to open the hole at the bottom of the WTS in case of a severe accident.

6.7.7 Plant layout

Various siting are possible for small reactors. JAERI has been studying concepts of nuclear barge (Ishida, et al., 2000), deep underground siting for supplying district heat at a city (Nakajima, et al., 2000), and a seaside pit siting besides a normal on-ground siting, available to PSRD.

The concept of seaside pit siting is as follows. The module reactors are set inside pits made in the bedrock at a seaside and submerged in seawater, that is, the outside of the containment vessel is filled with seawater. This siting allows unlimited cooling water - seawater - as the final heat sink to be provided. This method has also advantages of being strong against earthquake and crash of a flying object.

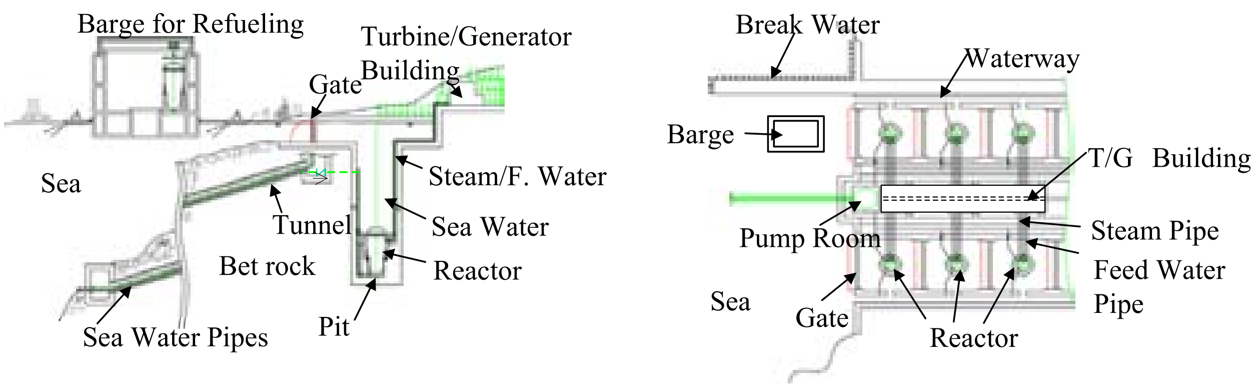


FIG. 6.7-7 Reactor siting in seaside pit

6.7.7.1 Buildings and structures, including plot plan

No information provided.

6.7.7.2 Reactor building

No information provided.

6.7.7.3 Containment

No information provided.

6.7.7.4 Turbine building

No information provided.

6.7.7.5 Other buildings

No information provided.

6.7.8 Technical data

General plant data

Reactor power output	31	MWe
Reactor thermal output	100	MWt
Power plant efficiency	31	net %
Cooling water temperature	15	C

Nuclear steam supply system

Number of coolant loops	1	
Primary circuit volume, including pressurizer	83	m^3
Steam flow rate	46.7	kg/s
Steam pressure	4.6	MPa
Steam temperature	289	C
Feed water flow rate	46.7	kg/s
Feed water temperature	185	C

Reactor coolant system

Primary coolant flow rate	450	kg/s
Reactor operating pressure	10	MPa
Coolant inlet temperature	270.4	C
Coolant outlet temperature	311	C
Mean temperature rise across core	41	C

Reactor core

Active core height	1500	mm
Equivalent core diameter	1620	mm
Heat transfer surface in the core		m^2
Fuel inventory	7,100	kg
Average linear heat rate	6.8	kW/m
Average fuel power density	15.2	kW/kg
Average fuel power density (vol.)	32.3	kW/l
Thermal heat flux	228	kW/m^2
Number of intermediate flow mixing grids	none	
Enrichment of first core	4.9	wt%
Operating cycle length	over 5	yrs
Average discharge burn up of fuel	26	GWd/t
Cladding tube material	Zr-4	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Active length of fuel rods	1500	mm
Burnable absorber	Gd_2O_3	in fuel rod
Number of control rods	37	
Absorber rods per control assembly	24	
Absorber material	B_4C	
Drive mechanism	Motor drive	
Positioning rate	300	mm/min
Soluble neutron absorber	NA	

Reactor pressure vessel

Cylindrical shell inner diameter	4000	mm
Wall thickness of cylindrical shell	237	mm
Total height	10800	mm
Base material: shell	SFVQ1A	
RPV head	SQV2A	
Liner	SUS	
Design pressure/temperature	11MPa/318 C	
Transport weight	283	t

Steam generator

Type	One through helical tube

Number	2	
Heat transfer surface	2540	m^2
Number of heat exchanger tubes	465	
Tube dimensions (inner/outer)	14.6/19	mm
Maximum outer diameter		mm
Total height	4000	mm
Transport weight		t
Shell and tube sheet material		
Tube material	Inconel 800	

<u>Reactor coolant pump</u>

Natural circulation, no pump

<u>Pressurizer</u>

Self pressurization
(saturated water pressure)

<u>Pressurizer relief tank</u>

Not applicable

<u>Primary containment</u>

Type	Water-filled RPV immersion	
Overall form	Cylindrical	
Dimensions (diameter/height)	7.3/13	m
Free volume	267	m^3
Design pressure/temperature	2 MPa/212C	
Design leakage rate	vol%/day	
Is secondary containment provided	No	

<u>Reactor auxiliary systems</u>

Reactor water cleanup	kg/s
Residual heat removal	kg/s
Coolant injection	kg/s

<u>Power supply systems</u>

Main transformer	kV
Plant transformer	kV
Standby transformer	kV
Start-up transformer	kV
Medium voltage busbars	kV
Number of voltage busbar system	
Voltage level of these	V dc
Number of battery-backed busbar	Vdc
Voltage level of these	Vac

<u>Turbine plant</u>

Number of turbines per reactor	1
Type of turbine	Steam turbine
Number of turbine section per unit	
Turbine speed	
Overall length of turbine unit	
Overall width of turbine unit	
HP inlet pressure/temperature	3.85/286 MPa/C

<u>Generator</u>

Type	3-phase, synchronous
Rated power	
Active power	
Voltage	
Frequency	50 or 60 Hz
Total generator mass	

Overall length of generator

Condenser

Type Rectangular. single flow
Number of tubes
Heat transfer area
Cooling water flow rate
Cooling water temperature
Condenser pressure

Condenser pumps

Number
Flow rate
Pump head
Temperature
Pump speed

Condenser cleanup system

Full flow/part flow
Filter type

Feed water tank

Volume
Pressure/temperature

Feedwater pumps

Number
Flow rate
Feed water temperature 185C
Pump speed

Condensate and feed-water heaters

Number of heating stages
Redundancies

6.7.9 Measures to enhance economy and maintainability

In design of the PSRD, high priority is laid on enhancement of safety as well as improvement of economy. For improvement of economy, simplification of the reactor system and long operation of the core are achieved. The economic measures taken in the PSRD are as follows.

- To simplify the design: The PSRD adopts an integrated-type PWR which has the integral type steam generator and the in-vessel type CRDM, but no circulation pump and no pressurizer in the primary loop (i.e. natural circulation and self pressurization). Neither are the volume control system, the purification system and the chemical shim system adopted. In the engineered safety system, the passive safety system with the water-filled containment are adopted, so that the safety injection pump and spray pump are not necessary. For the core decay heat removal in case of accidents, the EDRS with hydraulic force actuating valves and the SGs are used, which do not require the emergency electricity generator. The number of valves in the engineered safety system can be greatly reduced.

- To improve the operation flexibility: The PSRD can smoothly respond to change of demand, e.g. the daily load change. Several modular PSRDs will be used according to an electricity demand, rather than only one PSRD. Refueling or maintenance of a reactor can be done in turn without causing a large impact on the electricity grid.

- To reduce the cost of equipment and structures: The system of PSRD is simplified and the amount of equipment is greatly reduced as mentioned above. Adoption of the in-vessel type control rod drive mechanism (or CRDM) can contribute to compactness of the reactor vessel and the containment. The number of pipes which penetrate the RPV wall are very few and these pipes are only for the steam line, the feed water line and the safety valve.

- To reduce the construction period: The modular PSRDs will be fabricated in an exclusive factory and transported to the site, where they will be set in a short time.

- To reduce the scope of maintenance during operation and outages: The system of PSRD is simplified, and the amount of equipment and the number of pipes penetrating the RPV are greatly reduced as mentioned above. Maintenance or periodical inspection of the in-vessel type CRDM can be done after removal of the RPV cover together with them, not necessary to be separated.

- To make the maintenance easier and with lower radiation exposure: The number of the systems and amount of equipment are greatly reduced. The work will be done by remote control in the excusive building, after removing the whole containment and the RPV from the place where the reactor is set.

- To increase the plant availability and load factor: The core is designed so as to operate for the long life of five years with full power without refueling. Optimization of core design concerning the burnable poison ensures the burn-up of 26GWd/t for low enriched UO_2 fuel rods. Appropriate selection of number of the reactor set for operation corresponding to the demand can increase the plant availability and load factor.

- To reduce the power generation cost: The power generation cost is dominated by the construction cost and the operation cost. The construction cost can be reduced by fabrication of the module reactors, simplification of the systems and equipments, and making the RPV and the containment compact. The operation cost can be reduced by the continuous long- term operation without refueling and reduction of operators with introducing an automatic control system. The PSRD dose not basically require a manual operation for mitigation in case of accidents, since the passive safety system will work for them.

6.7.10 Project status and planned schedule

Conceptual design study of PSRD started in 2001 on the base of basic design of the MRX. Development of key component - the in-vessel type control rod drive mechanism- has completed in FY 2002. Design of PSRD with thermal power 300 MW for generating electricity has been conducting together with economy study.

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Appendix 1

THE GENERATION-IV SYSTEMS

Gas-Cooled Fast Reactor System (GFR)

The reference GFR is a 288 MW(e) system with a fast-neutron spectrum, helium cooled core with an outlet temperature of 850°C using a direct Brayton cycle. The plant is predicted to achieve 48% thermal efficiency. The system employs a closed fuel cycle for good utilization of fertile uranium resources and management of actinides through full actinide recycle, with co-located fuel cycle facilities.

Different fuels will be examined for their ability to operate with a fast spectrum at high temperatures with sufficient fission product retention. These are: advanced coated fuel particles, composite ceramic fuel, and ceramic clad elements of actinide compounds. The fuel element forms being examined are prismatic blocks, and fuel pin or fuel plate type fuel assemblies.

The GFR and its associated fuel cycle facilities are expected to be applied for electricity production and actinide management, and for high temperature applications (hydrogen production).

Lead-Cooled Fast Reactor System (LFR)

The LFR has a fast-neutron spectrum core, with a Pb or Pb-Bi eutectic coolant. This system utilizes a closed fuel cycle for good utilization of fertile uranium resources and management of actinides through full actinide recycle. Central or regional fuel cycle facilities are foreseen.

Plant designs under consideration have a range of plant ratings, including a long life (e.g. 15 to 20 years) "*battery"* core (a factory fabricated reactor module to be shipped to the plant site for re-fuelling) of 50-150 MW(e); a modular design of 300-400 MW(e); and a large 1200 MW(e) plant.

The *"battery"* reactor would be cooled by natural convection with an outlet temperature of 550°C, and possibly up to 800°C, depending on the success of the fuel and materials R&D. Foreseen applications include generation of electricity, hydrogen production and sea-water desalination.

Molten Salt Reactor System (MSR)

The MSR has an epi-thermal neutron spectrum with a circulating molten salt fuel (mixture of zirconium, sodium and uranium fluorides) and a full actinide recycle fuel cycle. Its reference power level is 1000 MW(e) with the primary system operating at low pressure (<0.5 MPa) with a coolant outlet temperature 700°C or above. The fuel flows through channels in the core graphite, which serves as a moderator. The heat is transferred through an intermediate heat exchanger to a secondary circuit and then through another heat exchanger to the power conversion system. Fuel processing is performed on line. Actinides and most fission products form fluorides in the liquid coolant and the fuel cycle can be tailored for efficient burnup of plutonium and minor actinides.

Sodium-Cooled Fast Reactor (SFR)

The SFR has a fast-neutron spectrum with sodium cooling and a core outlet temperature of 550°C. The fuel cycle is closed to achieve efficient utilization of uranium resources, with full actinide recycle. Plant options include: a 150 to 500 MW(e) system with metal alloy fuel; and a 500 to 1500 MW(e) system with MOX fuel. The core heat is transferred through an intermediate sodium-to-sodium heat exchanger to a steam generating system for electricity generation.

Supercritical Water Cooled Reactor (SCWR)

The SCWR is a water cooled reactor operating above the critical point of water (22.1 MPa, 374°C). The reference plant is 1700 MW(e) operating at 25 MPa with a core outlet temperature of 510°C to achieve a thermal efficiency of about 44%. There are two options: a thermal neutron spectrum reactor with once-through UO_2 fuel; and a fast neutron spectrum reactor with actinide recycle. Both options employ passive safety features similar to those of the simplified BWR.

For the thermal option, due to the low density of supercritical water, additional water is provided in the core in special “moderator rods”. The system operates in a direct cycle with no phase change. The balance of plant is based on supercritical turbine technology currently utilized in advanced fossil powered plants.

Very-High-Temperature Reactor System (VHTR)

The VHTR is a thermal neutron spectrum, graphite moderated, helium-cooled reactor with coolant outlet temperature of 1000°C, which enables high temperature process heat applications such as hydrogen production and process heat for the petro-chemical industry. The fuel is in the form of coated particles that can be in prismatic block or pebble elements. The reference concept is a 600 MW(th) plant with once through uranium fuelling supplying heat to the process heat plant through an intermediate heat exchanger. The system may employ electricity generating equipment for co-generation. The core can also be fuelled with U/Pu fuel.

Appendix 2

APPLICATION OF PASSIVE SYSTEMS

The application of passive safety systems [1], i.e. those whose operation takes advantage of natural forces such as conection and gravity, is potentially a significant means of achieving simplification and competitive economics in new nuclear power plant designs. The use of passive systems is not entirely new, and is not unique to any particular line of new reactor designs. But an increased reliance on this approach without diverse and redundant active backup systems, making safety functions less dependent on active components like pumps and diesel generators, may be an important means of achieving cost reductions for future plants.

Utility requirements documents that have guided design and development of future water-cooled reactors address the use of passive systems. For example, the EPRI ALWR Utility Requirements Document presents requirements for large ALWRs, having power ratings of 1200-1300 MW(e), and for mid-size (i.e. reference size of 600 MW(e)) 'passive ALWRs' that employ primarily passive means for essential safety functions. The European Utility Requirements (EUR) aim at next generation plants including those with passive safety features. The policy of the European utilities is to derive the maximum benefit from past experience with LWRs; however, the utilities are willing to consider passive safety features.

The IAEA Conference on 'The Safety of Nuclear Power: Strategy for the Future' [2] 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.

Some new water-cooled reactor designs rely on active systems of proven high reliability to meet safety requirements. Other designs rely on passive systems, while others rely on combinations of the two. The subject has been co-operatively reviewed by experts from several countries with their common views presented in a paper entitled "Balancing passive and active systems for evolutionary water-cooled reactors" in Ref. [4]. The experts note that designers consider first the fulfilment of the required safety function with sufficient reliability but must also consider other aspects such as the impact on plant operation, design simplicity and costs. The best effect for the plant safety may be achieved with a reasonable combination of active and passive systems to assure a certain safety function. Such combined usage can provide a decrease in the sensitivity of the safety functions to common cause failure, an increase in the plant safety and at the same time an improvement in economic performance. Key to cost reduction is elimination of safety functions requiring active safety support systems such as AC power, cooling water systems, heating, ventilation and air-conditioning systems, and the associated seismic buildings needed to house these systems and components.

The effects of passive and/or active safety systems on the overall plant safety can be quantified through the use of PSA methodology, yielding the values of the CDF and the LERF. Also, the effect of passive systems and inherent features in the 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 LERF 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.

Passive systems can be advantageous whenever such systems can provide one or more of the following benefits:

- Elimination of need for the short-term operator actions during accidents taken into account in the design;
- Minimization 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, radiation exposure, and in-service testing, due to reduction in the number of components and design simplification.

Thus, the choice of passive and/or active safety systems is based on the detailed consideration of their effect on the overall plant safety and total cost. In general, the most essential advantages of the passive systems are:

- Passive systems do not depend upon external energy supply;
- Passive systems simplify the safety system configuration and reduce the number of components;
- 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 systems include the lower driving forces and less operational flexibility. Due to low driving forces, the operation of these systems may be adversely affected by small variations in thermal-hydraulic conditions. The lower driving forces can also lead to the need for quite large equipment, and this factor may reduce the cost savings projected from elimination or downsizing of active components. 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 some cases, sufficient operating experience of the passive system/component under real plant conditions does not exist; so time-and-money-consuming research and development may be needed.

The design decisions with regard to the use of passive systems may also depend upon the functions assigned to the system. In particular, a system having an important role in the mitigation of severe accident consequences that is located in potentially contaminated areas (e.g., the part of the containment cooling system located inside the containment) could be designed to be 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.

The IAEA has organized several meetings to provide a forum of discussion on feasibility, technical issues, reliability, and development of passive safety systems [5,6]. These meetings have identified a number of issues regarding passive safety systems:

The quantification of reliability over a wide range of conditions, from severe accidents to normal operation, is key for the safety case and licensing and for defining requirements to be placed on other parts of the system. In the absence of a large database of relevant experience, methods must be established to determine reliability of passive systems:

- Their ability to operate sufficiently fast should be confirmed;
- They must not significantly degrade the operational performance of the reactor;
- Ageing of passive systems must be considered for the life of the equipment (e.g. to 60 years or more). Stored energy devices could degrade. Corrosion and deposits on heat exchanger surfaces could impair performance. The ability to demonstrate operational readiness over the life of the plant should be provided by appropriate in-service inspection and testing;
- Innovative components or features are likely to need extensive demonstration of technical feasibility. The ability to demonstrate operational readiness over the life of the plant should be provided by appropriate in-service inspection and testing, or a very strong case should be made that this is not required;

- In cases where passive systems are to be used together with active systems, or with active components, the economics of the combined system should be closely examined, considering that the active components will need back up power, operational diversity and redundancy; and
- Passive systems must be designed for ease of maintenance, and minimization of personnel radiation exposure.

Future international co-operation on the following topics may be useful:

- Initiation and reliability of passive systems;
- Testing and analysis of component and system performance;
- Quantification of uncertainties in computer codes; and
- Testing to address additional thermo-hydraulic phenomena that are being incorporated into these codes.

Testing of heat removal safety systems at large scale integral test facilities should continue to provide an extensive experience base in system behaviour and data for validation of computer codes used to predict performance of passive systems. While proof of predicted performance to satisfy safety requirements has been a major activity in support of design certification of plants with passive systems, further understanding of the basic heat transfer phenomena would be very worthwhile. This is especially important for passive heat transport systems which rely on small driving forces at low pressure thereby requiring comprehensive testing to assure that conditions resulting in system initiation and conditions affecting system reliability are thoroughly understood.

The need for additional research is very dependent on the specific reactor system design. Some examples for which research is underway include:

- Initiation of passive systems;
- Low velocity natural circulation;
- Effects of non-condensable gases on steam condensation;
- Water circulation in pools; and
- Rapid condensation caused by interfacing steam and sub-cooled water.

Code benchmarking activities on an international level are useful to assure proper modeling of passive components and systems.

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