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Improving economics and safety of water cooled reactors

Proven means and new approaches

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FOREWORD

Nuclear power plants (NPPs) with water cooled reactors [either light water reactors (LWRs) or heavy water reactors (HWRs)] constitute the large majority of the currently operating plants. Water cooled reactors can make a significant contribution to meeting future energy needs, to reducing greenhouse gas emissions, and to energy security if they can compete economically with fossil alternatives, while continuing to achieve a very high level of safety.

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.

The Department of Nuclear Energy of the IAEA is examining the competitiveness of nuclear power and the means for improving its economics. The objective of this TECDOC is to emphasize the need, and to identify approaches, for new nuclear plants with water cooled reactors to achieve competitiveness while maintaining high levels of safety. The cost reduction methods discussed herein can be implemented into plant designs that are currently under development as well as into designs that may be developed in the longer term. Many of the approaches discussed also generally apply to other reactor types (e.g. gas cooled and liquid metal cooled reactors). To achieve the largest possible cost reductions, proven means for reducing costs must be fully implemented, and new approaches described in this document should be developed and implemented. These new approaches include development of advanced technologies, increased use of risk-informed methods for evaluating the safety benefit of design features, and international consensus regarding commonly acceptable safety requirements that would facilitate development of standardized designs which can be built in several countries without major re-design efforts.

This publication has been prepared to address a recommendation of the IAEA Symposium on Evolutionary Water Cooled Reactors: Strategic Issues, Technologies and Economic Viability (Seoul, Republic of Korea, December 1998): namely that increased emphasis should be placed on achieving simplified water cooled reactor designs with improved economics. The task was carried out during 1999–2001 jointly by the Nuclear Power Technology Development Section, Division of Nuclear Power, and the Planning and Economic Studies Section of the Department of Nuclear Energy, in co-operation with the Division of Nuclear Installation Safety of the Department of Nuclear Safety. This report has been developed with participation of representatives from eleven industrial organizations and four government agencies as well as the OECD-NEA and the European Commission.

The IAEA appreciates the support of the following group of consultants who provided guidance and input for planning and preparing this TECDOC: E. Price (Canada); M. Vidard, Chairman, and J. Planté (France); T. Pedersen (Sweden); J. Board (United Kingdom); and G. Davis and R. Hagen (United States of America). The technical officers responsible for this publication were L. Langlois and J. Cleveland of the Division of Nuclear Power, working in co-operation with A. Gomez-Cobo, M. Gasparini, and F. Niehaus of the Division of Nuclear Installation Safety.

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1. INTRODUCTION

Most of the world's electricity markets are moving towards greater competition. Both private sector and state-owned electricity generating organizations must be increasingly concerned with the cost of their operations, and must focus on supply technologies that are low cost and low risk.

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.

Electricity is being sold into competitive power grids at less than 3 US cents/kW·h in a number of countries. Even in traditionally protected electricity markets, target power generation costs are falling, and are expected to continue to fall as reforms continue. To approach economic competitiveness under these market conditions, the base "overnight" capital costs of nuclear plants (not including interest, inflation and escalation during construction; contingencies, major refurbishments or decommissioning costs) would have to be reduced by approximately 1/3 from previous cost estimates, to the range of US $900 to $1400/kW(e), depending on the country being considered.

Therefore major capital cost reduction through simplified designs and shorter construction times must be achieved, together with continued reductions in fuel and operating costs. Regulatory procedures and requirements must be stable and predictable, and must not require expensive features that provide only insignificant reduction of risk already below stringent regulatory criteria. Cost-effective design measures for meeting these requirements must be found. Development of technological improvements and streamlining of regulatory requirements must go hand in hand to reduce both the capital and the operating costs of nuclear plants.

Approaches that are being implemented or explored for reducing costs, include:

- application of technological advances in design, and construction (e.g. computer aided design processes to establish efficient modular construction techniques);
- modern digital instrumentation and control systems;
- components with built-in diagnostics to achieve high reliability with less redundancy;
- design for higher temperature (higher thermal efficiency);
- design for multiple applications (e.g. co-generation of electricity and heat; sea water desalination);

— reduction of number of components and materials requiring nuclear grade standards;
— application of passive safety systems; and
— a move to more risk-informed safety regulation.

This TECDOC examines economic factors influencing the competitiveness of nuclear energy, and outlines viable approaches for achieving more cost effective designs of future plants while maintaining high levels of safety.

2. THE NEW CONTEXT — ECONOMY, NEED AND MARKET

2.1. THE STATUS OF NUCLEAR POWER PLANT COMPETITIVENESS

2.1.1. Existing plants

While support for nuclear energy has waxed and waned over the past several decades, many operating nuclear power plants are proving to be valuable assets in competitive markets. These plants reliably generate electricity at competitive costs, with minimal environmental impact, and without creating undue risks to the general public. Nuclear plants are also recognized as making important contributions to security of energy supply, especially for countries that import significant quantities of coal and oil.

Growing competition in electricity markets is leading to major changes in the structure of the electric power industry. The most significant change is that electric power is no longer seen as a "natural monopoly" in which a single supplier provides electricity to a regulated but protected market with essentially guaranteed revenues, and where cost and financial risk management are not vital concerns. Private sector participation and market liberalization are fostering a competitive marketplace for power generation, even if transmission and distribution are often left as regulated functions. Prices will increasingly be set by the market rather than by regulation, and suppliers are increasingly forced to focus on efficient and profitable operations.

Some industry experts feared that more competition would lead to the demise of the current fleet of nuclear plants. To the contrary, however, it has produced a nuclear industry that is more competitive than it has been in decades. Preparing for the transition to a competitive environment has forced all parties — including regulators, plant owners, and suppliers — to look closely at the economics of nuclear power relative to alternative technologies for electricity generation.

Existing nuclear plants have, in recent years, achieved substantial improvements in performance (e.g., plant availability) and reduced operating costs (e.g., reduced staffing levels), and nuclear fuel costs have remained low. As a result, many older and largely amortised nuclear units are operating in cost ranges that are very competitive with the electricity production costs of natural gas and coal plants. In 1999 US nuclear power production costs (fuel plus operation and maintenance costs) dropped to an average of 1.83 US cents/kW·h, compared to 2.07 cents/kW·h for coal-fired plants, 3.18 cents/kW·h for oil-fired plants, and 3.52 cents/kW·h for natural gas-fired plants [1]. In 1999 the most efficient nuclear plants in the USA achieved production costs of 1.1 cents per kilowatt-hour.

Decreasing costs are also reported in France where the cost of a kilowatt-hour (including amortization, fuel and operation and maintenance costs) generated with nuclear power by Electricite de France in 2000 was between 15 and 18 French centimes depending on site. This is a decrease of 7% since 1998 [2]. Comparisons of generation costs in France in 1997 (before the recent price increases in fossil fuels), reported by Framatome at the 9th International Conference on Nuclear Engineering, show nuclear generation costs as being the lowest[1]. Information provided by the European Commission based on experience of Bayernwerk AG (Germany) shows that in 1998 production costs from largely depreciated nuclear and hard coal plants were 1.57 to 1.88 €cents/kW·h for nuclear and 2.08 to 2.38 €cents/kW·h for coal; and representative costs for new gas-fired plants were 2.49 to 2.69 €cents/kW·h. Moreover, the marginal costs for nuclear power are usually less, and are certainly less volatile, than marginal costs for gas- or coal-fired plants because of the relatively lower share of fuel costs in the overall cost of nuclear generation. In a competitive market environment a grid operator would buy power from the lowest marginal cost supplier; hence nuclear power may often have a dispatching advantage.

These improvements in production costs are being helped by the stabilization of the safety regulatory environment in the last few years. After more than a decade of changes mandated by the safety regulators because of accidents at Three Mile Island and Chernobyl, the regulatory environment has improved significantly. For example, recognizing that regulatory stability is a critical component in assuring economic competitiveness of the existing nuclear plants, the U.S. Nuclear Regulatory Commission is adopting a risk-informed, performance-based regulatory structure that is proving to be highly effective and efficient. On top of this, the plant owners are obtaining regulatory approval for extending the lifetimes of their operating units. As a result, there is a sense of optimism that the nuclear industry can be viable for decades to come.

Moreover, there is a growing concern about the effects that greenhouse gases will have on the earth's environment. While acceptance of nuclear power within the environmental community is still uncertain, it is indisputable that nuclear power contributes a major share of achieved greenhouse gas reductions. With more than a quarter century of safe operation, coupled with its clean air benefits and cost stability, nuclear power has the potential to make an increasing contribution to the world's energy needs while contributing to greenhouse gas reductions.

In Europe, the extensive challenge of ensuring security of energy supply while being confronted by increasing external dependence and the urgency of fighting against climate change, is the basis for the recent decision of the European Commission to launch a debate on future European energy strategy with its "Green Paper" on energy supply security.

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[1] The specific values reported by Framatome, based on information from the French Ministry of Industry, for a base production of 6000 hr/year, and a discount rate of 8%, were:

nuclear:	20.8 French centimes (fuel: 4.5; O&M: 3.37; investment: 12.7)
coal:	3.5 French centimes (fuel: 10.1; O&M: 4.5; investment: 8.9)
gas:	22.6 French centimes (fuel: 15.5; O&M: 2.2; investment: 4.9)
wind:	30 - 50 French centimes (investment: 25-40; O&M: 5-10)
solar:	200 - 300 French centimes.

As a result of all of these factors, there is support for extending the lifetimes of the existing nuclear units in some countries. There are several reasons why lifetime extension of successful nuclear plants can be profitable. Their debt is largely amortised, they have a revenue stream, operating costs are already low, and decommissioning fund obligations are nearly satisfied. Compared with the cost of building new plants, investment costs for lifetime extension, while not trivial, are likely to be lower because costs such as civil works, land acquisition and site preparation are not incurred, and significantly less equipment must be purchased. Life extension can also be attractive for environmental reasons where compliance with air pollution standards or commitments to greenhouse gas emissions reductions argue against increased fossil fuel fired generation.

Plant up rating is also an economically attractive option that has already been accomplished at many existing plants. Plant up rating, even without lifetime extension, effectively adds new capacity and therefore reduces unit costs. Up ratings of 10–20% have been achieved at many plants. Modern turbines for water cooled reactors are now more efficient than those installed in the 1970s and 80s (34% vs 30% efficiency) and upgrading a turbine can result in 3 to 5% more power being delivered to the grid.

For plants that would require extensive upgrades to qualify for license extension, the economics of the decision should be carefully evaluated. If the operating organization cannot afford the required upgrades, or if these entail investment costs that cannot be recovered profitably over the projected remaining life of the plant, then life extension is not an option and the plant should close. It is worth noting, however, that shutting down a plant or cancelling a construction project is also potentially expensive, as most contracts have cancellation costs or penalties for early termination. Again the economics of the case must be carefully weighed. Even where closure is politically motivated or the result of policy decisions, transparency in government would require an assessment of the economic consequences of the decision.

Appendix 1 provides further discussion of technologies for extending plant lifetime, improving plant availability and reducing operation and maintenance costs based on information provided at some recent IAEA technical meetings.

2.1.2. Projecting costs of generating electricity for new plants

During the 1980s and 1990s, considerable development efforts were conducted for new water cooled reactors of advanced designs [3] and [4]. A large part of this effort was on evolutionary LWRs and HWRs incorporating the large base of design, construction, licensing and operating experience of existing plants together with several technological developments for improving performance and safety [5]. Regulatory requirements and industry standards that had continually evolved since the introduction of nuclear power were adopted in the design bases.

Since these efforts began, the cost target for commercial success has been decreasing, as electricity prices have tumbled. In 1995, a generation cost of US$ 0.043 per kW·h was considered the goal for new nuclear power plants to be competitive in the USA. By 1998, the target had dropped to US$ 0.03 per kW·h [6]. The Electric Power Research Institute included projections for future generating costs in its “Electricity Technology Roadmap”. Those projections showed the base estimates for electricity generated from coal or natural gas to be less than US$ 0.03 per kW·h, by the year 2020 (in 1998 dollars). In July 2000, the U.S. Department of Energy included the 3-cent generation cost target as a tentative economic goal for development of new reactors in its report "Discussion on Goals for Generation IV Nuclear Power Systems".

The competition has not been standing still. Significant improvements have been made in the thermal efficiency of coal and especially gas-fired electricity generating plants, setting new economic standards that nuclear power has to meet. The thermal efficiency of gas-fired plants has risen to well over fifty percent and is expected to reach sixty percent before the end of this decade, even without the further efficiencies that may be gained through co-generation. This compares with a thermal efficiency for water cooled nuclear power plants that is in the mid-thirty percent range. Moreover, these gas-fired power plants have relatively short (less than 2 years) construction times. It is not surprising that natural gas technologies can dominate the new power generation market, particularly in times of low gas prices, driving out not only nuclear but in many instances coal as well.

By contrast, the nuclear industry is struggling to maintain its market share. Since the mid-1990s, there have been construction starts and / or grid connections in the Middle East (India, Islamic Republic of Iran, and Pakistan), the Far East (China, Japan, and the Republic of Korea), in Latin America (Brazil, and Mexico), and in Eastern Europe (the Czech Republic, Romania, and the Slovak Republic). However no new[2] nuclear power plant construction has started in the United States, Canada, and Western Europe since 1977, 1985 and 1991 respectively.

To capture economies of scale, nuclear plants tend to be of larger capacity than coal and gas plants. They also have a higher capital cost per kW(e), and can historically take up to 10 years from project initiation to commercial operation. This results in the need for considerably larger amounts of capital to be provided for financing new nuclear projects relative to new fossil projects. In developing countries the financing problem is compounded by OECD investment rules that add a 1% risk premium to lending rates on all OECD export credits where nuclear power plants are concerned. The financing difficulties associated with high capital cost certainly played a key role in the postponement in July 2000 of the Akkuyu project for a first nuclear power plant in Turkey, although Turkey is not subject to the 1% risk premium. Postponement of the project is attributed to the fact that the Government of Turkey could not afford the estimated 3 to 4 billion US dollars needed to finance the project (see Annex 1). The long construction times of new nuclear plants also poses the risk that during construction the costs, regulations, policies and markets may all change, jeopardising both completion and a return on investment. Can such risks and costs be reduced or secured sufficiently for nuclear power plants to successfully compete for financing in capital markets?

Studies on projected costs of generating electricity provide results that depend strongly on the assumptions used. However, what such studies do underline is that cost management and flexibility will always be required to meet market uncertainties. Moreover, given the range of market conditions and generating costs, and the wide variety of assumptions used to forecast such costs, no single technology can be declared optimal in all markets or countries.

As noted in the beginning of this section, cost targets change with market conditions, and the nuclear industry needs to be responsive to changing market conditions in order to be competitive. Assuming that new nuclear plants can achieve electricity production costs of US$ 0.01 per kW·h (consistent with the best of current experience of existing plants — see

[2] There are possibilities to re-open a closed unit and to resume construction of some partially completed units.

Section 2.1.1), meeting the generating cost target of US$ 0.03 per kW·h mentioned above requires that new plants achieve overnight capital costs[3] of US$ 900–1000/kW(e) for an example privately financed project (i.e. a discount rate of 11 percent, operating life of 20 years), or US $ 1300–1500/kW(e) for an example publicly financed project (i.e. a discount rate of 8 percent, operating life of 40 years)[4].

Other insights can be obtained by examining results of the study [7] that was carried out by the OECD/IEA-NEA, in co-operation with the IAEA. This study provides one set of estimates of capital cost and power generation costs for nuclear, coal and gas fired plants in several countries, assuming a commissioning date of 2005, an economic lifetime of 40 years, a load factor of 75% for all plant types. As shown in Table I, the base overnight capital cost for new nuclear power plants around the world, including some evolutionary water cooled reactor designs under development at that time, were projected in that report to range from US $1,440 to 2,260 per kW(e) installed for 80 percent of the cases. Since the cost values shown in Table I depend on a series of assumptions specific to the study, they should be considered as indicative only. It should be noted that the costs projected by the study do not include the cost of risks that affect a project's credit rating, such as non-completion, exchange rate fluctuations and cost over-runs.

TABLE I. CAPITAL COSTS AND CONSTRUCTION TIMES FOR DIFFERENT ELECTRICITY GENERATING OPTIONS(1) (SOURCE: OECD, 1998)

	Total capital cost per kWe installed2) US $ 80% of cases (total range from the OECD report)	**Base cost3) per kWe US $ 80% of cases**	**Construction period Years**	**Typical unit size MWe**
Nuclear Water-cooled reactors	**2,070 - 2670 (1690 - 3150)**	**1440 - 2260**	**5 - 8**	**600 – 1,500**
Coal	**1160 - 2020 (1050 –2930)**	**840 - 1550**	**4– 5**	**400 – 1,000**
Natural gas CCGT	**510 – 970 (450 – 1770)**	**420 - 810**	**1.5 - 3**	**250 — 750**

1 Costs were transmitted by the participating countries to the OECD expressed in national currencies of 1 July 1996 and were converted to US dollars at the exchange rates of that date.
2 including interest during construction (IDC), contingency and cost of major refurbishment for an assumed 10% discount rate.
3 "overnight cost" without IDC, contingency and cost of major refurbishment.

3 "overnight capital cost" is the capital cost without including interest during construction, contingency and costs of major refurbishments.
4 These examples assume a plant capacity factor of 85%, a construction time of 60 months, and a capital amortisation period of 20 years.

Important assumed parameters affecting power generation cost comparisons include the nuclear plant capacity factor and plant lifetime. As was mentioned above, the OECD/IEA-NEA study assumed a nuclear plant capacity factor of 75% and a plant lifetime of 40 years. A higher value for the nuclear plant capacity factor would be more favourable for nuclear power. A plant lifetime of 60 years would also be somewhat more favourable for nuclear power. However, the nuclear option would appear less viable for an assumed 20-year amortization period, more typical of private market conditions in which investors want a rapid return on investment. The assumed price of gas is also very important — being both volatile and country and geographically specific, its value does not everywhere put the same pressure on nuclear generating cost. In the OECD-NEA report, no single technology (coal, gas or nuclear) is projected to provide the lowest generation costs in all countries analyzed. In cases where the power generation costs with gas were projected to be as low as US $ 0.03 /kW·h, the report showed that drastic reductions of nuclear plant capital costs would be required in order to restore the competitiveness of nuclear power.

Thus the projected viability of the nuclear option depends on the specific market conditions and cost assumptions used in the cost analyses — all of which may vary from country to country. Importantly, in addition to economics, a country's national policy issues, such as diversity and security of its energy supply, may affect the decision on whether or not to construct nuclear power plants.

The importance of country specific assumptions is illustrated by more recent projections of generating costs, such as those for new base-load power production in Finland (see Annex 2). In these studies, projected generating costs were compared for nuclear power, CCGT, coal-fired and peat-fired plants. Assumptions included a capacity factor of 91 percent (which is justified for nuclear on the basis of experience with the Olkiluoto units of Teollisuuden Voima Oy, Finland) and a discount rate of 5 percent. The results were that nuclear power was predicted to provide the lowest generating costs.

Annex 3 addresses the cost economics necessary for nuclear units to be competitive based on results of a series of OECD studies on projected costs of generating electricity and related activities.

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[5].

2.2. RESULTING IMPLICATIONS

The competitiveness of nuclear power with gas (or coal) power is being challenged in some countries more than in others. Nonetheless in all countries capital cost reductions are needed to secure or enhance the competitiveness of nuclear power plants. Therefore, it is necessary to

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[5] 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 lower than those for coal fired plants and CCGT plants [33].

fully implement proven means of cost reduction, and to examine and implement new approaches, as described in section 3. Although the needed capital cost reduction varies from country to country, the results of section 2.1.2 generally suggest that a 30% reduction is an appropriate goal.

Further reductions in power production costs (O&M and fuel) should also be pursued. However, these costs have already been significantly reduced for existing plants in a drive to become more competitive. Because of these efforts, power production costs could approach 1.0 US cent /kW·h, as it has for the most efficient plants in the USA, and future nuclear plants are unlikely to see these costs drop significantly below this, without major innovations.

Significant reductions in power generation costs of existing plants have also been achieved by increasing the availability factor. Efficiently managed plants are now achieving availability factors of 90% and above, so higher availabilities, and the associated economic gains, for new plants will not be large.

Electricity markets reflect growing emphasis on profit and rewarded risk, on cost and risk management. In this respect, financial analyses and the use of financial criteria such as net present value and internal rate of return as well as pay back period, will be more useful than traditional cost comparisons for discerning sensitivities and the potential profitability of future investments in power plants in more competitive markets[6].

Because new nuclear power plants carry high financial risks and perceived uncertainties, they must offer high returns to attract investors. To achieve this in competitive markets capital costs of new nuclear plants will need to be significantly reduced. This is the main commercial challenge facing designers and manufacturers of new plants, and it is a prerequisite to a revival of the nuclear power option.

3. APPROACHES TO REDUCE NEW PLANT COSTS

3.1. PROVEN MEANS TO REDUCE CAPITAL COSTS

There is a common set of approaches for reducing costs during any construction project, including nuclear projects. Several studies [8] [9] [10] 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 schedule;
4. Standardization, and construction in series;
5. Multiple unit construction;
6. Simplifying plant design, improving plant arrangement, and use of modelling;
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.

[6] The IAEA has developed a number of models to facilitate such analyses, including MESSAGE, FINPLAN, GTMAX, and BIDEVAL.

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. In some countries, **economies of scale** are being pursued for new, large evolutionary plants. For example, in the Republic of Korea, the development of the Korean Next Generation Reactor (KNGR) was started in 1992, building on the experience of the 1000 MW(e) Korean Standard Nuclear Plants (KSNPs) that are now operating and under construction[7]. Recent development focused on improving availability and reducing costs (see Annex 4 and Ref. [11]). A power level of 1400 MW(e) has been selected to capture economy of scale. In March 2001, KEPCO started the Shin-kori 3,4 project for the APR1400 (Advanced Power Reactor 1400), and announced that APR1400 is the new name for the KNGR considering the start phase of the Shin-kori project.

In Japan, in 1991 a development programme for ABWR-II was started, aiming to further improve and evolve the ABWR[8]. A target of 30% reduction in power generation cost from that of a standardized ABWR has been set. In order to gain the benefits of economies of scale, the electric power for ABWR-II has been increased to 1700 MW(e), relative to 1360 MW(e) for Japan's first two ABWRs (Kashiwazaki-Kariwa Units 6 & 7). Commissioning of the first ABWR-II is foreseen in the late 2010s (see Annex 5).

In Europe, the developers of the 1545 MW(e) European Pressurized Water Reactor (EPR) have adopted a higher power than the latest series of PWRs operating in France (the N4 series) and Germany (the Konvoi series) to capture economies of scale. An optimization of the conventional island design (see Annex 6) resulted in a net thermal efficiency of 36% contributing to the increase in plant electrical capacity. Improving NPP conventional island design with a focus on lowering capital investment costs is highly important, as this part of the plant can constitute 30 to 40 % of the total capital cost. Annex 6 identifies several innovations and improvements incorporated into the design of the conventional island of the EPR which are expected to have a direct impact on lowering investment costs. Framatome ANP estimates that the generation costs over the lifetime of a series unit for EPR will be approximately 18 French centimes.

In the USA, efforts are currently underway by Westinghouse for a 1090 MW(e) plant called the "AP-1000" building on the passive safety technology developed for the AP-600, which received design certification from the U.S. Nuclear Regulatory Commission in December 1999. Westinghouse estimates that this scale-up will reduce the base overnight capital costs to US$ 900–1000 / kW(e) (see Annex 7) to meet the target of 3.0 cents / kW·h mentioned in section 2.1.2, with an assumed 20-year financing at a "commercial" rate-of-return.

Economies of scale are also pursued in Russia by Atomenergoprojekt and Gidropress, which have begun development of the WWER-1500 (see Annex 8). The developers claim that this plant will have a capital cost per kW(e) that is 10 % less than that for the WWER-1000 (V-

[7] The first two KSNPs, Ulchin 3 and 4, have been in commercial operation since 1998 and 1999 respectively, and four more units (Yonggwang 5 and 6 and Ulchin 5 and 6) are under construction.

[8] The first two ABWRs in Japan, Kashiwazaki-Kariwa 6 and 7, 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, and under licensing at Ohma Unit 1. Another eight ABWR plants are in the planning stage in Japan.

392) design, of which two units are planned to be constructed at the Novovoronezh site, and 40% less than currently operating WWER-1000 (V-320) units. The cost of power production with the WWER-1500 is predicted by the developers to be approximately half of the average cost of power production from current NPPs with operating WWER-1000 (V-320) units.

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 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.

The CAREM reactor development programme in Argentina provides an example of development of an SMR with simplified design features. These features include an integrated primary system with natural circulation (elimination of primary system piping and pumps), self-pressurization (eliminating the pressurizer) and passive safety systems. A next step of the project is to construct a prototype of about 27 MW(e) (see Annex 9). Following construction of a prototype, a scale-up to a somewhat higher power level would help to bring the cost into the competitive range.

The approach (see Annex 10) taken by AECL in Canada to develop next generation CANDU plants 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 CANDU-NG (described in Ref. [12]) which is scheduled to be available after 2005, incorporates design modifications based on use of slightly enriched fuel, a modified fuel bundle design, and light water coolant at somewhat higher temperatures and pressures. The design is more compact and achieves a somewhat higher thermal efficiency than current CANDU plants. Other improvements include large scale modularization, prefabrication and the use of 3-D computer modelling to optimize the design and construction process. AECL claims that the overnight cost target of US $ 1,000/kW(e) with a construction time of 48 months can be achieved for the Nth unit.

Reducing the construction schedule 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. An example of successful application of “large block” modular construction is that used in Japan in the construction of the Kashiwazaki-Kariwa Unit 7 [5]. Other methods that have proved efficient in Canada, Sweden and Brazil include open-top access, slip-forming, parallel construction, and sequencing of contractors. Use of computer models to schedule engineering and equipment delivery, modularization and open top construction techniques are being implemented by AECL for the Qinshan-3 project. Also crucial is the efficiency of construction management, which requires a close customer-vendor working relationship. Besides these changes in construction method, other measures that could reduce the construction schedule include advanced engineering methods, and up-front engineering and licensing. Often ignored but also important are delivery cost improvements. An option with considerable potential for cost savings is delivery of a plant by barge. There was such a plan in Russia for the KLT40 reactor and it was investigated extensively in Canada for the CANDU 300 plant. Savings of 30 to 40% on construction time were predicted by the developers.

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 has resulted 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, including the advantages and the inconveniences of standardization, is provided in Ref. [5], in the paper by B. Roche of EdF.

In the Republic of Korea, the benefits of standardization and construction in series are being realized with the KSNP units. Accumulated experience is now being used by KEPCO to develop the improved Korea Standard Nuclear Plant, the KSNP$^+$ (see Annex 4).

Benefits of standardization and construction in series are also being realized in Japan with the ABWR units. 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 [13]. With regard to development of ABWR-II, TEPCO expects that the first unit of this series will cost less than the last ABWR, just as the ABWR first-of-a-kind was less expensive than the last unit of the 1100 MW(e) BWR series. The eventual goal is that after building, for example, 10 ABWR-II units, a capital investment savings of 20–30% relative to the last ABWRs would be realized (see Annex 11).

Closely related is the cost-saving practice of **multiple unit construction** on a single site. Experience reported by several countries shows that 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. An example of a multiple-unit CANDU project is the Wolsong four unit station, in the Republic of Korea which, as a result of multiple unit construction, achieved capital cost reduction in siting, land preparation for the transmission system, licensing for identical units, site labour, and common facilities such as administration and maintenance buildings, warehouses, roads and guard stations.

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. Some of these are discussed in greater detail in section 3.2. There is now broad support within the industry and among some of its regulators to move toward a risk-informed, performance based regulatory process, whereby the regulator establishes basic requirements and sets overall performance goals, while plant management decides how best to meet the stated goals [14]. A basic vehicle for this approach is the probabilistic safety assessment/probabilistic risk assessment (PSA/PRA).

An approach that has been adopted in the USA is new licensing process (provided for in the Energy Policy Act of 1992) for future nuclear plants which is intended to be a more predictable process with less uncertainty, and less financial risk to the applicant. The current plants were licensed under a system that had two major steps: the construction permit, and the operating license. Utilities received construction permits based on preliminary designs, so potential safety issues could not be fully resolved until the plant was built. Delays sometimes resulted from changing regulatory requirements during construction. Under the new licensing procedure, a utility could select a design that has been certified by the U.S. NRC and for which safety issues related to the design have been solved. The U.S. NRC would address site suitability before construction starts. After holding one or more public hearings, and if all regulations are met, the U.S. NRC may issue a combined construction and operating license.

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, labour costs — may result. The China National Nuclear Corporation (CNNC), in developing the CNP-1000 plant, is pursuing self-reliance both in designing the plant to meet Chinese safety requirements, and in fostering local equipment manufacture, among other measures, and they are doing so with a view to reducing construction and operation costs for these new nuclear power plants. Other means, such as higher plant availability, reduced construction times, standardization, and incorporating lessons learned from the design, construction and operation of Qinshan and Daya Bay NPPs, are also being applied. (see Annex 12).

In India, a continuing process of evolution of HWR design has been carried out since the Rajasthan 1 and 2 projects (see Annex 13). Means to reduce cost include standardization of design and commercially available equipment, use of optimum versus best grade material, compact layout, reduction of number of welds by use of custom built piping. Cost control during construction has been achieved through freezing the bulk of the design before start of construction, modular and parallel construction, equipment vendor standardization and pre-qualification of vendors. The continuity of the programme and a shift from manual to mechanized construction have also contributed to construction cost control. Optimization of the schedule for equipment ordering has led to reductions in financing costs during construction. Contributing to cost control, remarkable reductions have been made during the Indian HWR programme in the time taken from criticality to synchronization with the grid, and from synchronization to commercial operation.

3.2. NEW APPROACHES TO REDUCE CAPITAL COST

The previous section discusses traditional proven approaches which should achieve cost competitiveness for nuclear power plants in some markets and countries. However, in other markets and countries where nuclear power faces very strong competition from fossil alternatives, these approaches may be insufficient to assure that it is the lowest cost option for generating electricity. As discussed in section 2, the nuclear community must continue to move forward in identifying new approaches for further reducing the costs of new nuclear plants.

In the course of nuclear power development in the latter part of the twentieth century, there have been significant developments in 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 has taken a prominent role in the discussion of the safety principles for future reactors and the safety goals they should achieve. These are addressed in several INSAG documents [15] [16] [17] [18]. Also the IAEA convened a conference in 1991 on The Safety of Nuclear Power: Strategy for the Future [19]. Subsequently, in 1995 the IAEA published TECDOC-801 — Development of Safety Principles for the Design of Future Nuclear Power Plants [20]. From these 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.

Evolutionary plant designs have incorporated many new features to achieve high performance and safety [5]. Individually these features have relatively small effect on plant capital cost. However, when taken collectively, they have resulted in a pronounced impact on the overall cost and economics of new nuclear plants. Hence, to meet the challenge of competition, and the resultant need for cost minimization, the economic impact of design choices must be thoroughly evaluated to assure that design requirements are cost-effective.

This re-evaluation cannot wait to be carried out in major increments, every decade or so (as the nuclear industry has done in the past), but must become a continual process that is constantly looking for ways to improve profitability. Otherwise, the rapidly changing marketplace could simply leave nuclear energy behind.

The following describes new approaches to reduce nuclear power costs which should be further developed and implemented. Section 3.2.1 discusses possible means of reducing plant cost through increased application of probabilistic safety analysis in design and licensing. It first provides some perspective on historical developments in the use of probabilistic safety analysis to assess nuclear plant safety. Following that is a discussion of potential approaches for achieving economic designs for future water cooled reactors by using probabilistic safety analysis to evaluate the cost effectiveness of the existing deterministic criteria – along with some examples of technical areas for which application of risk-informed approaches for design and licensing may lead to cost savings for future designs through simplification. Section 3.2.2 addresses development of advanced technologies that can complement, and in some cases support, design simplification achieved through risk-informed approaches to design and licensing. Section 3.2.3 addresses the application of passive safety systems. Section 3.2.4 identifies some potential areas that may result in cost savings through a re-evaluation of user design requirements with a focus on economic competitiveness. Section 3.2.5 discusses improving the technology base for eliminating over-design, and Section 3.2.6 discusses the advantages that could be gained by achieving international consensus regarding commonly acceptable safety requirements that would facilitate development of standardized designs which could be built in several countries.

3.2.1. Increased application of PSA in design and licensing

3.2.1.1. Historical perspectives on developments in probabilistic safety analysis and criteria

Probabilistic safety analysis (PSA) methodology is used in many industries for identifying failure scenarios and deriving numerical estimates of risk. Reference [21] provides an historical perspective of the use of PSA for nuclear plant design and in the regulatory process for nuclear plants.

The nuclear plant licensing process in several countries has historically been based on deterministic regulatory requirements. Plant design and operational requirements have been derived through the analysis of Design Basis Accidents (DBAs), selected to envelop credible accident conditions, supplemented by the single failure criterion.

Some elements of probabilistic reasoning have influenced the development of these deterministic requirements. For example, plants are not required to be designed against multiple, simultaneous, independent failure events (accident initiating events) considered to be each of low probability. Similarly, meeting safety criteria with multiple, independent safety system failures is not a design requirement.

Reliance on regulations based on deterministic criteria has historically led to overlooking potentially significant safety issues. This was demonstrated by the accidents at Three Mile Island Unit 2 and Brown’s Ferry. Probabilistic safety analyses have shown that there are risks of accidents resulting from events that occur outside of the design basis, and are due to multiple failures, human errors and external events. The realization that PSA provides a framework for addressing uncertainties and for providing significant insights into contributors to risk, led to an increased acceptance of PSA as a regulatory tool to “backup” the deterministic requirements. This also highlights a major benefit of PSA — namely that it provides a very useful base both for establishing regulations and for optimizing designs because it can examine the impact on safety of incorporation of new technologies and changes in design.

For nuclear power plants, PSA is useful in providing insights about plant design by identifying dominant risk contributors and comparing options for reducing risk. Analyses can be conducted at three levels:

— Level 1 — which assesses plant design and operation, focusing on sequences that could lead to core damage. This is quite useful in examining the design strengths and weaknesses for prevention of core damage;
— Level 2 — which addresses core damage, the response of the containment, and the transport of radioactive material from the damaged core to the environment. The analyses show the relative importance of events, and allow investigation of measures for mitigating the consequences of accidents; and
— Level 3 — which analyses dispersion of radio-nuclides into the environment and potential environmental and health effects.

Use of PSA in the safety assessment process requires establishment 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). It is not the purpose of this document to suggest or recommend PSC values. The following is a brief summary of some considerations regarding PSC.

Different types of PSC can be considered. PSC can be related to the *core damage frequency*[9] (CDF), which is the most common measure of risk. CDF is predicted by performing a Level 1 PSA, and the majority of plant operators have performed Level 1 PSAs. 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 the *large early release frequency* 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).

IAEA Safety Series No. 106 [22] suggested a PSC framework utilizing both a design *target* / *objective*, and an upper *limit*: the upper limit (or zone) representing the threshold of unacceptability, and a lower level representing the design target / objective. Between these two levels, all reasonably practicable measures should be taken to reduce risk; below the target level there would be no pressure from the regulatory body to reduce risk, although the designer or operator may choose to do so. Safety Series No. 106 further pointed out that with such a framework, it is not necessary to define separate probabilistic safety criteria for current and new plants. Current plants would be expected to be somewhere between the two levels, while new plants would be near or below the level of the design target. This two-tier framework has been formally adopted in the United Kingdom [23], and the basic idea is in more widespread use.

The best known PSC related to CDF is the set of objectives put forth in INSAG-3 in 1988, and re-affirmed in 1999 in INSAG-12: a CDF of 1×10^{-4} for existing plants, and a CDF of

[9] Core damage is defined as resulting from accidents involving loss of adequate cooling (either due to an under-cooling or over-power event) to reactor fuel elements up to and including major damage to a reactor with internal release of fission products, but not necessarily involving a release into the environment (loss of containment integrity).

1×10^{-5} for future designs[10]. These are currently applied formally and informally. For example, in the United Kingdom, 1×10^{-4} is a formal limit for current plants, and 1×10^{-5} is regarded as the objective for old and new plants. These CDF values are expected to be based on best estimate analyses. In the US advanced LWR programme, 1×10^{-5} was a design criterion, but for purposes of investment protection, not as a safety criterion.

With regard to a core damage PSC, IAEA Safety Series No. 106 proposes a *target* frequency of 10^{-5} per reactor-year, with no single accident sequence contributing a significant percentage of the target.

With regard to a large off-site-release PSC, IAEA Safety Series No. 106 notes three examples (in the United States of America, France and the United Kingdom) of practice in Member States in which an *objective*, *target* or *goal* of a frequency of 10^{-6} has been adopted, and suggests that, until such time as an international consensus has been reached, the target frequency for a large off-site release should be 10^{-6} per reactor year, and that this criterion should be used in conjunction with the existing concepts of defence in depth and diversity, with no one accident sequence contributing a significant percentage of the risk.

IAEA Safety Series No. 106 also points out that limitations in PSA have in the past resulted in placing reliance on deterministic procedures, techniques and criteria to ensure safety, with PSA and PSC playing a complementary role. However, it goes on to suggest that the evolution of deterministic regulatory guidelines should take into account insights from PSA to identify areas where the coverage of existing regulations may be inadequate, as well as areas where regulations may be overly stringent.

[10] Discussions of PSC *targets* for CDF and large off-site-release have been provided for more than a decade in documents of INSAG [15–18] which serves as a forum for exchange of information in nuclear safety issues of international significance. 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”.

The use of probabilistic targets as part of design safety criteria has developed in the UK over a long period. By the time the Sizewell B project was started in the mid-1980s, the use of PSA was widely used for the U.K.'s advanced gas cooled reactors. Its extensive use in design and operation for Sizewell B provides a useful case study on PSA as an important tool in decision making (see Annex 14).

The regulatory approach taken in Argentina provides an example in which probabilistic and deterministic approaches are used in a complementary manner for assessing the safety of a given installation or operation. Two decades ago the Argentine Regulatory Body adopted a probabilistic criterion of risk acceptance while keeping additionally some deterministic requirements. A plant-specific PSA of Atucha Nuclear Power Plant (CNA-I) at its level-I included the analysis of the internal plant events that could occur during full power operation as the initial condition, and was finished in March 1996 and then improved in up-dated versions. The results of the study showed some weaknesses in the plant design and operation which resulted in requirements made by the Regulatory Body in order to perform corrective actions. The fulfilment of these and other improvements contributed effectively to increasing the safety level of CNA-I. For the other Argentina operating NPP, Embalse Nuclear Power Plant (CNE), the level I PSA is also applied. For new designs, such as those of innovative reactors, PSA will be used both by designers as a design feedback tool and by the regulatory authority in the licensing process.

For existing plants in the USA there is growing use of PSA in regulation. This is referred to as *risk-informed regulation*. The general objective of risk-informed regulation is to focus regulatory attention in a manner that is consistent with the risk importance of the equipment, events and procedures to which the requirements apply, so that both regulatory and licensee resources are efficiently used in ensuring public safety. This objective implies that the requirements be commensurate with the risk contribution. Risk-informed regulatory criteria are expected to result in a systematic means for efficiently expending resources for achieving an overall balance in safety of nuclear power plants. Furthermore, application of risk-informed criteria provide quantitative means of assessing compliance with regulations.

In the USA, recently issued regulatory guides for operating plants deal with risk-informed in-service inspection, risk-informed graded quality assurance, risk-informed technical specifications and risk-informed in-service testing. Several generating companies are applying these guides with expectations of significant cost savings. For example, risk-informed graded quality assurance is expected to yield savings from reducing expenditures for quality assurance for systems, structures and components that are insignificant from the risk point of view.

In Japan, operators also look forward to improved economics through strategies including regulatory change based on operational experiences and increased use of risk analysis methodology. For example, graded quality assurance based on risk insight is considered by TEPCO as one means of reducing costs (see Annex 11).

In Canada, the licensing approach has risk-based origins, and the licensing framework is non-prescriptive. The regulator sets the safety goals and requirements which the designer must meet, but does not prescribe how to meet them. The result has been flexibility in the development process for new designs. Through a process called "up-front" licensing, agreement is reached by the regulator and the designer on how the requirements will be met, before issuance of a construction permit. PSA and cost-benefit analyses play an important role in "up-front" licensing as a way of providing an objective framework for decisions. It is noted however, that prescriptive deterministic requirements may be more detailed, but are not necessarily more demanding from a safety point of view, than probabilistic requirements (see Annex 15).

In the design process for evolutionary water cooled reactors (e.g. System 80+, the ABWR, the EPR, the KNGR, and others), PSA has been used to supplement deterministic design requirements. PSA has been used in the design process, for example, to identify and resolve plant vulnerabilities, intersystem dependencies and potential common cause failures, examine the risk benefits of different design options, and examine the balance between preventive and mitigative measures. The PSA for a final design may be submitted to the regulatory body as part of the supporting documentation.

3.2.1.2. Achieving economic designs for future water cooled reactors using a risk-informed design and regulatory process

In the past, PSA has been used primarily to identify weaknesses in plant designs. Only rarely has PSA been used to assess the usefulness of deterministic criteria that already exist. As noted in the previous section, in the USA the risk-informed regulatory approach is now starting to be used to re-evaluate criteria that affect operating nuclear plants.

To help to assure that future plants are cost competitive, there is a need for significant risk-informed modifications to the regulatory approaches for the licensing and oversight of reactors. In this regard, Annex 16 discusses trends and needs in regulatory approaches for future reactors.

Today, the state of the art for PSA (including a large database of operating experience) is sufficiently mature that it could be used to help to identify the safety requirements that unnecessarily complicate the design without significantly contributing to the system's or structure's overall reliability. Features incorporated to satisfy deterministic requirements that add to plant costs without adding real or significant safety benefits could be reconsidered. To simplify the designs, the new process could use probabilistic safety assessment as a design tool to determine how each system or structure could be simplified, while maintaining a reliability level that ensures safety. Such a process would be most effective if combined with the introduction of smart equipment (see section 3.2.2.2.), which would increase reliability at the component level.

Several industry experts consider that nuclear plant simplification would result from development and application of a more risk-based design and regulatory process, backed up where necessary by deterministic requirements, rather than re-evaluation of all of the existing deterministic criteria, one-by-one. In this approach, the design features must meet the functional requirements (control of reactivity, removal of heat from the core, confinement of radioactive materials and control of operational discharges, as well as limitation of accidental releases), but the way in which these functional requirements are met would be justified by using PSA to demonstrate that both the design is well balanced with respect to safety and that the PSC are met. This approach would incorporate improved PSA methods to address issues such as human performance and probabilistic analysis uncertainties as well as improved technology such as stronger materials and "smart" equipment with health monitoring systems. Deterministic requirements would still be anticipated, but to a lesser extent than is currently the case, and with the primary purpose of backing up the risk based requirements where uncertainties in the PSA methods are considered to be sufficiently large to warrant deterministic requirements[11]. The expectation is that requirements would be much less prescriptive and would provide plant designers with much greater flexibility to simplify the plant designs.

Annex 17 presents the status and some results of a project managed by Westinghouse and funded by the U.S. DOE's Nuclear Energy Research Initiative which is investigating the feasibility of a risk-based approach to design and licensing.

Risk-informed decision making can play an important role in development and optimization of future reactors through simplification of safety systems and a sound safety classification of structures, systems and components. A challenge will be to sufficiently establish PSA tools, including understanding of the uncertainties in predicted results, to demonstrate that sufficient defence in depth, and sufficient balance among the various levels of defence in depth, can be achieved through simpler and cheaper technical solutions [24] [25].

Establishment of a risk-informed regulatory system would require considerable work including further development of PSA[12], establishment of applicable PSC by regulatory bodies, translation into inspection and acceptance criteria for plant operation and maintenance, as well as means of incorporating other regulatory objectives (e.g. limitations on worker exposure, latent effects to a larger population, land contamination, etc).

The following documents, recently published by the IAEA, together with Safety Series No. 106 [22] and INSAG-12 [18], could serve as very useful sources of the more functional (and less prescriptive) deterministic criteria that would still be retained, in the more risk-based regulatory process:

— Safety Fundamentals: The Safety of Nuclear Installations; Safety Standards Series No. 110; and
— The Safety of Nuclear Power Plants: Design Requirements; Safety Standards Series No. NS-R-1.

The document "Safety Fundamentals: The Safety of Nuclear Installations" presents basic objectives, concepts and principles of nuclear safety. The document "The Safety of Nuclear Power Plants: Design Requirements" compiles nuclear safety requirements applicable to safety functions and the associated structures, systems and components as well as to procedures important to safety. It is recognized that technology and scientific knowledge will continue to develop, and that nuclear safety is not a static entity; however, these requirements

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[11] This implies that the PSA assessments must include quantification of the uncertainties (both aleatory [i.e. random] and epistemic [i.e. dependent on the degree of validation]), and that the PSC must include indication of unacceptable uncertainty.

[12] As discussed in Annex 16, a viable risk-informed regulatory system will be highly reliant on PSAs and their bottom line results. Before this is tenable, there must be consensus on what constitutes an acceptable quality PSA. Improvements to PSAs will be required, including

(a) the ability to assess the total risk including shutdown, low power, fires and external events;
(b) a methodology for including into PSAs safety culture and organizational factors, as well as better treatment of human factors;
(c) a methodology to better include ageing effects;
(d) the ability to incorporate a full uncertainty assessment; and
(e) improved methodology to examine the risk importance of simultaneous reliability changes of large numbers of components not normally explicitly modelled in PSA.

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.

Examples of opportunities for cost reduction

The following provides some examples suggested by the industry for which use of PSA to help to identify the safety requirements that unnecessarily complicate the design without significantly contributing to the system's or structure's overall reliability may result in cost savings. These are presented as worthy of more in depth examination. They do, nevertheless, provide an indication of potential changes in plant design and system simplification that might be made to reduce cost while maintaining very high safety levels.

3.2.1.2.1 Simplification of the emergency core cooling system of PWRs

Reactor Cooling System (RCS) piping is designed to withstand mechanical and thermal loads resulting from all operating conditions contemplated during plant life. The conservatisms and margins embedded in the design process give confidence that RCS piping has an extremely low-probability-of-failure. However, since the early days of the nuclear industry, the instantaneous, double-ended, cold leg guillotine break loss-of-coolant accident (LOCA) has been the reference primary system pipe-break accident for the design of PWRs in the USA[13].

By the early 1980s, the U.S. NRC acknowledged that ductile pipe would "leak before break" and could not pose a real threat — as long as there was a leakage detection system. On this basis, the NRC at that time allowed "leak before break" to be credited, in satisfying some *new* NRC requirements; however, the double-ended guillotine pipe break was maintained as the basis for already established regulatory requirements — which had served as the design basis for several safety systems including the safety injection system (SIS) and the containment system. Maintaining these requirements as the design basis for several safety systems was justified by the regulator, and accepted by industry, as providing an added safety cushion, to cover the unknown. In a young industry — lacking a wealth of operating experience and data — an added safety cushion, to cover the unknown, was not unreasonable. Now, with the accumulated experience of the industry, a re-examination of these requirements is considered by the industry to be warranted (see Ref. [6] and Annex 17).

Designing against an instantaneous double-ended cold leg guillotine break LOCA, and compliance with ECCS acceptance criteria which require use of a set of penalizing assumptions (including among others the assumption that only safety-related components and systems function), leads to a Safety Injection System (SIS) design with:

[13] The U.S. Code of Federal Regulations TITLE 10: ENERGY; CHAPTER I--NUCLEAR REGULATORY COMMISSION, PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES states that in analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system. The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe.

— an injection capability well in excess of what would be needed in case of LOCA (even for double-ended guillotine breaks),
— multiple trains (at least two) to address break location and the single active failure criterion;
— accumulators (provided to address re-flood in the early phase of the accident until the safety injection pumps can provide adequate cooling water); and
— fast-starting, high unit capacity diesel generators (because Loss of Offsite Power may occur simultaneously with LOCA).

If risk-informed requirements were established with more realistic assumptions on pipe break size and with more realistic assumptions for accident analyses, it is possible that the SIS could be greatly simplified. In an example case provided by Westinghouse (see Ref. [6] and Annex 17), the following design simplifications and modifications are contemplated[14]:

— elimination of the instantaneous double-ended guillotine break as a credible accident;
— elimination of the accumulators to supply water before the start of the injection pumps;
— a decrease of injection pump capacity requirements; and
— elimination of the requirement for fast-starting, high capacity diesel generators, allowing use of more standard Diesel generators, which have higher reliability.

Such modifications in the design approach could relieve loadings to be considered for designing other structures, or provide more flexibility for layout optimization. This would decrease the number of components to be inspected, periodically tested, and would reduce radiation exposure to plant staff.

To go further, designers could even specify smarter pumps and valves that provide more diagnostics and warnings to the operators — increasing the reliability of each safety injection train (by improving reliability at the component level). Combined with more realistic design bases, it is possible that designers could develop a system with fewer redundant trains than found in current designs — with each train being simpler than those in the current designs. In the example analysis provided in Annex 17, it was suggested that most of the safety injection equipment could be deleted, with a cost savings in the range of US $15–20 million. Further cost savings would result because elimination of any SIS trains would reduce the costs of the structures required to house the SIS, as well as the other systems that support it. It is also likely that designers could then simplify the system's operating, maintenance, and testing procedures.

In a further step based on PSA, non-safety-grade components could be relied on according to their *probabilities* of failure including consideration of their inspection and maintenance, rather than simply *postulating* failure of these systems[15]. An example of approaches that could be reconsidered would then be the single failure criterion since adequate injection capability,

[14] Further analyses would need to consider all accident sequences (e.g. also steam line break) that influence the design of the safety injection system, and would need to examine the uncertainty distribution in the PSA results so that confidence levels can be addressed.
[15] Components rated as "non-safety related" are assumed not to function in the safety analyses. However, such components can perform safety functions, and they are highly reliable. When they are considered in probabilistic safety analyses, the results show that the evolution of the significant parameters during a LOCA are much milder than deterministic analyses, which assume they do not function, predict.

which could meet probabilistic safety criteria, based, for example, on CDF or on LERF, could be provided with low-probability-of-failure, non-safety grade components.

3.2.1.2.2. Simplification of the containment design

The containment system of a reactor plant represents a major component of the cost and schedule for constructing the plant, and both parameters are strongly influenced by the requirements on its physical strength and integrity. The prime function of the containment structure is to confine the radioactive products that may be released from the reactor system in the event of an accident and prevent their transport to the environment. To this end, the containment has to be designed to accommodate the highest pressure and temperature conditions that may arise following accidents "within the design basis". Rupture of the largest diameter pipe of the reactor coolant pressure boundary (pipes connected to the reactor primary system) has traditionally been taken as the design basis.

In the 1980s, regulatory requirements were increased to consider containment capabilities and functions for "beyond design basis" accidents, including core melt situations. Core melts would result in pressures well above the normal design values due to release of hydrogen by oxidation of some of the core materials. In many countries, containment venting (or pressure relief) systems with filters were introduced to protect the containment structure and to ensure a controlled release path. In the 1990s, requirements developed further in some countries: the containment should withstand the maximum pressure that might result from a core melt with oxidation of all core materials for a certain time period — often one day or 12 hours — without activation of any pressure relief system.

To meet these requirements, containment structures have had to be made larger and stronger; larger to limit the pressure increase, and stronger to raise the "ultimate strength" capability. The ultimate strength refers to the pressure and temperature that the structures can withstand without losing their integrity. The structural stresses will exceed the normal limits set by codes, but the strains and stresses in the containment liner and in the pressure-retaining steel structure must remain safely below the ultimate strain capabilities of the respective elements. For pre-stressed containment structures the ultimate strength would typically be in the order of twice the design pressure. Considering the extremely low probability of severe accidents, design requirements based on ultimate strength capability criteria with adequate margin rather than on criteria set by the extreme pressures of core melt situations is expected to be more cost efficient in assuring that the containment structures will maintain their integrity.

Given the extremely low probability of double ended guillotine pipe breaks and severe accident events, a careful examination of the containment design requirements, using the insights of probabilistic risk analyses, may identify more cost effective design solutions which simplify the containment design while still controlling the consequences of accidents including severe accidents.

An estimate of cost savings and reduction in the construction duration achieved by simplifying the containment design for the KNGR[16] is given in Annex 4.

[16] Now named the APR-1400

3.2.1.2.3. Re-evaluation of equipment qualification profiles

Under the current approach, only components and systems qualified as safety-grade can be used in the deterministic safety assessment. Safety-grade equipment must satisfy a number of special conditions – e.g., quality assurance, environmental qualification, and seismic qualification, in a more stringent way than other equipment. In particular, they must be able to perform their intended functions under the harsh environmental conditions to which they would be exposed during a design basis accident (e.g. LOCA or earthquake). A qualification profile is thus established and only those components that pass the qualification tests are considered adequate for use in nuclear plants.

Significant margins are embedded in the qualification profile. For example, the environmental profile for those components relied upon in case of a break in the reactor coolant piping is based on:

- a "reasonably enveloping scenario" requiring the designer to postulate that a significant number of claddings have failed as a result of the accident;
- a "reasonably bounding source term", meaning that the amount of fission products assumed to be released from the core to the RCS first, and then to the containment, is conservative compared to currently available data[17].
- an additional margin (typically of about 10%) on the calculated radioactivity intended to account for normal variations on commercial production of equipment and errors in defining satisfactory performance (IEEE Std. 323-1974); and
- a conservative evaluation of exposure time.

A re-evaluation of the qualification profiles for safety-grade components that closely examines the embedded margins, and the need for these margins, may result in simplification of the design of some components.

3.2.1.2.4. Reduction of the number of components and materials requiring special nuclear quality standards

Experience shows that the traditional practice of developing special "nuclear" quality specifications for components that are classified as safety-grade equipment results in prices that are significantly higher than for corresponding components manufactured to commercial standards; a price increase factor of 1.5 is not uncommon, and in particular cases may reach 3 or more. Further studies in this area could identify components, for which nuclear quality specifications are currently required, that could be manufactured with sufficient reliability using industrial standards. However, component reliabilities would need to be examined over the range of conditions for which they must operate, including, for some components, severe accident conditions.

Some typical cost and quality impacts are due to the following factors. The special "nuclear" specifications typically include stringent material specifications that deviate from the manufacturer's standard, and that obviously add to cost. Also important are the requirements for inspections and acceptability reviews during manufacturing, that lead to disturbances and delays in the process flow and a slower manufacturing process.

[17] In the USA, the "reasonably bounding source term" is based on information in the document "Accident Source Terms for Light-Water Nuclear Power Plants" (NUREG 1465).

In many cases, the reason for developing detailed “nuclear” specifications have been the need for establishing more stringent quality assurance and quality control procedures, driven by regulatory requirements. Today, corresponding rules and procedures are an integral part of the normal international standards that manufacturers follow for other industries with demanding service conditions, however, and the need for “special treatment” for use in nuclear plants should be rather limited, or non-existent.

A change to less prescriptive requirements would obviously result in significant cost reductions. Negative consequences such as impaired component quality and reliability are expected to be negligible. On the other hand, nuclear specifications may still be found necessary for special equipment — particularly for equipment of significant safety importance, e.g., the reactor coolant pressure boundary. This leads to a suggestion for adopting a safety classification into three categories A, B and C, respectively, in similarity with the categorization of IEC 1226 for electrical equipment. Category A would encompass components, systems and structures (CSSs) of which the function has major safety importance, B would comprise CSSs of less importance, while C would deal with CSSs related to operation. Special “nuclear” specifications should be limited to Category A. Category B should primarily be subjected to well established codes and standards, while conventional equipment and standards are presumed for Category C. In the context of reducing costs as far as practical, it will obviously be necessary to minimize the amount of CSSs to be included in Category A. To examine the risk impact of such a safety classification system, further development of PSA methodology would be required, because standard PSAs do not cover the large number of components of interest.

Instrumentation and Control (I&C) systems for new plants provide a good example where savings could result by reducing the number of components requiring special nuclear standards. I&C systems will beyond doubt be digitized systems, and there are strong arguments for building these from readily available “commercial off the shelf” components, taking advantage of the provenness of these in other industrial applications. In this way, a high reliability of the hardware and basic software can be ensured, and then the major verification and validation efforts can be concentrated on the application software — or the plant adaptation. The new technology offers a host of advantages for control and supervision of the plant and its equipment, and it can provide access to almost unlimited amount of information, and also a discrimination of information so that the operators will not be confused by numerous simultaneous alarms when something happens.

Experience shows that modern software of digitized I&C systems generally is more reliable than that attained by the earlier hardware-based I&C systems. Still, some regulators express distrust in the modern systems, and require diversified digitized or direct hard-wired equipment as backup for the most important safety functions.

Annex 18 presents results of a study carried out by EdF and Framatome regarding means of reducing the purchasing cost of nuclear power plant components while maintaining an equivalent level of safety. Results showed numerous feasible opportunities for using good non-nuclear industrial practice in areas of general instructions (quality assurance, supplier surveillance and documentation) and technical requirements. Savings of at least 10% for the purchasing costs of equipment for the nuclear island in a series plant were identified.

3.2.1.2.5. Improved inspection and maintenance practices based on risk-informed analyses

A key factor ensuring high capacity factors and operation with high margins of safety from power plant operation is effective maintenance. Plant surveys by regulatory and advisory agencies show that high capacity factors and high safety levels are predictably linked.

Effective maintenance is one factor of excellent economic management of a nuclear generating plant. In new plants efficient maintenance should be a result of good design. From operating experience to date, sufficient information on material selection and expected operating environments is now available to predict material behaviour and reduce unexpected maintenance to very low levels. This should also be possible with the use of non-nuclear standard materials in many applications, as discussed in the previous section. Well-engineered processes and procedures can overcome limitations of deficient designs, but future designs also need to better accommodate the needs of maintainers with respect to space/access to equipment and systems for inspection and repair, meeting the requirements of the relevant worker protection code. Attention is also needed in the design of equipment to facilitate adjustments, repair, rehabilitation or if necessary, replacement. It is now generally recognized that the most efficient way to do repairs on a component is to remove it from the system in which it is located and replace it with a spare. Repairs can then be done in a workshop environment instead of in the constrained environment of the plant that is under pressure to get back on line. This implies modular construction as well as the existence of an operative spares policy for both new and existing plants that should apply in particular to valves, pumps and other maintenance intensive components. For new plants, there should be a standardization policy with respect to bolts, head screws, etc that limits the sizes and thread types over various systems to as small a number as possible. A further feature of new designs should be a greater capability to do in-service testing and preventive or corrective maintenance at power. This will be effective in reducing the workload during shutdowns.

The bases of effective maintenance are well-engineered procedures and processes for operators, comprehensive training for maintainers, and life management plans for major components. Such plans should now be based on component performance and risk-informed analyses, from which can be derived the necessary intervals for in-service inspections or in service testing to ascertain component or system condition. Such risk-informed analyses should avoid the unnecessary work burdens and radiation exposures imposed by too frequent inspections. Too frequent inspections also incur the risk of malfunctions due to human errors in the repair operation or re-assembly. The inspection and maintenance intervals for less important components (from a safety consequence point of view) should be less rigorous and based on the performance of similar components in similar environments. In effect, maintenance should now be based on the extant operating experience, not on engineering judgement and should not be a prescribed activity. This follows the trends in U.S. NRC regulatory practice where PSA-based operational activity is a permitted procedure for guiding operations.

The existence of well-engineered procedures is particularly necessary when maintenance practice is done as a contracted-out task. The planning for maintenance must still be done in-house, and the diligence with which planning is done will determine the efficiency of the maintenance activities.

The life cycle planning for inspection and required maintenance is critical to the economics of plant operation since it impacts directly into lost generation from extended shutdown duration and to various degrees from the cost of replacement components resulting from inadequate maintenance or control of operating conditions. The development of a computerized maintenance management system, as described in section 3.2.2.2, is an essential part of any reliability management programme because it makes available not only historical data associated with a management application, but should be part of a broader planning tool, integrating financial factors, resource planning etc., to achieve continual updates on plant health.

One measure of the effectiveness of maintenance of a plant is the ratio of scheduled maintenance to unexpected maintenance undertaken during a shutdown. While agreement on the exact ratio may vary from organization to organization, the ratio should be high enough that it does not impose a significant economic penalty on the operator. Nominally a ratio higher than 5:1 would be desirable.

Where adequate historical component or system behaviour is not available or where backup information is desirable, the incorporation of sophisticated monitoring devices ("smart" technologies) can provide the necessary guidance to the operators, in direct measurements or analysed trends as described in Section 3.2.2.2.

Another possible avenue of maintenance activity that could be explored is to emulate the aircraft industry in allowing the vendors to provide lifetime maintenance or service contracts to operators for the lifetime of the plant. This would require the vendors to arrange sub-contractors to do the work supervised and controlled with technical specifications and procedural/process documents that are kept up to date by the vendor based on experiences and developments over a number of plants.

3.2.2. Development of advanced technologies

Advanced technologies, (especially those involving computer-based applications) have increased productivity and efficiency in high technology industries such as the aerospace and automobile industries. The nuclear industry has also benefited from such technologies but not to the same extent, partly because the rise in use of such technologies has occurred when the nuclear industry has been in a period of stagnation with respect to new orders; the industry has seldom been able to exploit the benefits of multiple orders, and the long regulatory review process of new modifications with the associated intensive safety analyses have offset the schedule advantage of their implementation. However, advances have been made in some areas and advanced technologies must be considered an essential element of any effort to increase the competitiveness of new nuclear generation. It is to be expected that the potential application of advanced technologies to achieve profound reductions in cost and delivery in the nuclear industry will be realized in two categories: (a) the design, procurement, manufacture and construction phases of a project where it can be tied to cost reduction from improved manufacturing schedules, and (b) in the operational phase, where it will have a large effect in enhancing safety and efficiency of operation. These two categories are discussed as follows.

3.2.2.1. Design, procurement, manufacture and construction

The effective and timely execution of nuclear plant design, procurement, manufacture, construction and maintenance activities, is highly dependent upon the flow of necessary information throughout the integrated project cycle, starting with design and continuing through to commissioning and operation and maintenance.

There are two challenges to be addressed. The first challenge is producing the design in the first place while maintaining configuration control. In order to drive capital costs down, the use of 3-D modelling is seen as the keystone in the process. The computer aided design process has now been used for design-related activities and is a standard feature of most design and graphics organizations. It allows the rapid exploration of design variations and modifications; the ability to see the impact of changed designs on the system layout, and in the case of modular systems, the ability to virtually see the accessibility of the module into the plant during construction or after operation.

The entire physical design of the plant should thus be rendered in a 3-D model that encompasses 100% of the detailed physical plant. In this visionary concept there are no drawings, or specifications, only the database that represents the model. All documents and graphics used to construct are “cut” from the model by area and construction trade as the plant is constructed.

The next step is to integrate the data and electronic tools (wiring and cabling, material management, equipment and document asset management) so that every piece of data exists in only one place and changes to the physical, logical and analysed plant cascade automatically through the data so that correctness and consistency are assured, thus making detailed configuration control throughout the plant life cycle feasible.

A natural extension of the 3D model is its integration with a schedule model. In this way, the implications of documentation or data changes can be easily assessed during construction.

One key to cost reduction has been reducing design and construction cycle time as shown by the automobile and aerospace industries. These industries have gone through the process of re-examining how their engineering, procurement and fabrication are accomplished and how information technology can be integrated into these processes. For example, concurrent engineering represents the potential to work a number of activities in a parallel path, thus shortening the schedule. Dramatic improvements will be possible once an understanding is established on how to take these concurrent engineering activities such as materials review and stress analyses and deal with resolving multiple revisions of the same drawing being used by multiple designers and disciplines.

The second challenge is getting the components and subassemblies designed, ordered and assembled without error. Co-ordination and communication between various organizations are needed. The Internet has now reached a high level of capability with respect to vendors supporting Web-enabled applications. This new technology can be exploited through the common application of integrated electronic concurrent engineering tools across the design to manufacturer interface, to achieve substantial reductions in the capital cost of major plant components like steam generators, reactivity mechanisms, pressurizers, pumps, etc., to enable the design and procurement cycle to be speeded up substantially.

A further natural development is the merger of supply chain management with network technology such that the reactor vendor, the architect engineer and the plant purchaser and all the various suppliers receive co-ordinated information to achieve improvements in practice.

The co-ordinated design information on components can be delivered in appropriate form to the component vendor without the need for successive co-ordination activities.

The evaluation of the cycle times outlined above will serve to highlight the schedule impacts of the design, fabrication and construction processes, but it will also facilitate evaluation of whether changes in the plant design will also reduce capital cost.

Capital cost reduction goals need to be established for selected high capital cost systems and structures in a plant's design. The quantitative targets of these goals can be established by (1) first reviewing the systems and structures in a plant's design; (2) making initial judgements about what the target reduction for the individual system or structure should be; (3) considering the relative cost significance of the system or structure; and (4) recognizing that the individual targets will need to be regularly re-evaluated and adjusted, considering changes occurring in the competing energy technologies (e.g. natural gas generated electricity), and re-adjusting the target, if necessary.

Critical reviews can evaluate how particular features impact on initial cost and schedule of typical nuclear plant designs. One could, for example, evaluate the principle that individual stand-alone units should have no shared facilities. On the one hand, this can result in unnecessary duplication of some service facilities. On the other hand, sharing of facilities can lead to safety questions of common mode failures and possible interaction between units. Other areas to be evaluated include modularization, construction techniques, and prefab/shop assembly. Critical reviews should be performed to challenge current assumptions and requirements on plant arrangement and design, and to propose alternate safety criteria to reduce the cost of nuclear power plants for the future while assuring that safety goals are met.

New technologies should also be evaluated for their cost-reducing potential. The results of these evaluations could then be fed back into the process evaluation outlined above to determine the overall impact and produce a process guideline document. This document would then be used by all of the individuals and organizations looking for ways to significantly simplify the systems and structures to reduce capital costs. This will prevent large disparities in accomplishments from one group to another and development of conflicting design requirements.

The above tools and the associated data used during design, procurement and construction could be transferred to the operations organization in such a way that there will be no need for the costly time consuming reconfiguration of data that occurred in the past. Detailed materials, equipment, document and database configuration management in operations will simply be the continuation of the configuration management maintained throughout design, procurement and construction.

Annex 19 describes the approach taken by AECL to integrate the design, procurement and construction processes. This system is currently being used for the Qinshan project in China, and it is planned to be used for future CANDU projects.

Annex 20 describes two programmes being carried out by Westinghouse and Duke Engineering and Services to develop advanced technologies to reduce design, procurement, construction, installation and testing (DPCIT) costs. The first programme, funded by the Electric Power Research Institute, is developing a 4-D model of construction plans for the

System 80+ design. The second programme, funded under the USDOE's Nuclear Energy Research Initiative, focuses on application of new technologies to reduce DPCIT costs.

3.2.2.2. Smart technologies to increase reliability and efficiency of operation

Safety is enhanced by the operational effectiveness of a well trained and conscientious work force provided with the tools that can analyse trends in component or system behaviour and diagnose situations. Such tools are predicted to result from the application of *smart* technologies that can monitor the health of systems and components and indicate the approach of conditions outside the design/operating envelope. In this way, high reliability can be achieved from the existence of such diagnostic capability, supporting a case for eliminating redundant equipment.

Current activities within the USDOE's Nuclear Energy Research Initiative include a programme to design, develop and evaluate an integrated set of tools and methodologies that can improve the reliability and safety of advanced nuclear power plants through the introduction of *smart* equipment and predictive maintenance technology (see Annexes 17 and 20). "Smart equipment" embodies elemental components (e.g. sensors, data transmission devices, computer hardware and software) that continuously monitor the state of health of the equipment in terms of failure modes and remaining useful life, to predict degradation and potential failure and inform the users of the need for maintenance or system-level operational adjustments. The combination of smart equipment and predictive maintenance technology would provide a system-level integration of plant maintenance information and real time sensor data utilizing the self-monitoring and self-diagnostic characteristics built into the equipment. The system could be designed around a distributed software architecture that allows scale up to enterprise-wide applications and provides the ability to view real time equipment performance and safety-related data from remote locations.

Internal network technology and high speed communications will make it possible for all of the engineering, analysis, licensing and procurement functions, that may now be the responsibility of individual operating stations, to be co-ordinated by a central engineering group serving many stations. The station staff complement will be significantly reduced, but safety should be enhanced by powerful plant health monitoring systems. The use of such a system implies the existence of suitable computer system protection or "firewall" to the system.

Future web-based technology will be the vehicle by which operations and engineering staff will locate, view, understand and "navigate" the work processes that guide their day to day tasks. When the system is completed and deployed, downtime and maintenance costs will be significantly reduced.

To facilitate the introduction of such technology into new reactors, it will be necessary to have available a preliminary evaluation of the existing reactor designs and their operation to produce data on what nuclear plant equipment would most likely benefit from the addition of *smart* features, identified and prioritized using available maintenance data and PRA studies. It could then be shown how *smart* features could be applied to a specific piece of equipment.

In detail it will likely be necessary to develop a methodology for evaluating plant equipment and systems to determine an optimum health monitoring plan. Critical equipment, dominant failure modes, and dominant failure causes will need to be identified and ranked.

Optimization analyses can then determine the most cost-effective allocation of *smart* features, together with an understanding of the adequacy of existing sensor technology, and where new sensor technology is needed. As mentioned above, such work should identify and use, whenever possible, existing or ongoing studies and/or analyses dealing with plant maintenance data, PRA studies, on-line component monitoring, and condition-based maintenance. On-going work in the area of condition-based maintenance, and in the area of predictive maintenance and optimal spares analysis, also form valuable inputs.

It may be necessary as a parallel exercise to develop methodologies for (1) systematically evaluating the equipment used in a nuclear plant to determine how the reliability of the equipment could be improved by the addition of more sophisticated (i.e. *smart*) monitoring and diagnostic features; and (2) designing plant systems that will allow communication and integration of data among the *smart* components, as well as the control room systems and the plant operators.

It is envisaged that an equipment maintenance reliability simulation ("virtual machine") capability will need to be developed. A recurring issue in demonstrating the benefits from *smart* equipment and predictive maintenance systems is the lack of documented data demonstrating the benefits. It is currently not possible to document cost savings and performance improvements from health monitoring systems since there are no completed plant installations currently collecting data. The lack of active installations also means that there is no test bed to help develop and test the analysis portions of a health monitoring system. For example, rules must be developed and tested for updating model data and for making "repair/do not repair" recommendations. The parameters that control these rules need to be optimized for the equipment behaviour and the quality of starting data. For these reasons, a means of simulating equipment behaviour is required.

It will be necessary to have a "vigilance" program to identify *smart* technologies available from industry/government programs that can be applied to new nuclear plants and identify technology gaps for which *smart* technologies do not currently exist and, therefore, must be developed.

In particular, methodologies for consolidating the presentation of data obtained from *smart* equipment to end-users will be needed. Health monitoring systems often require the processing and analysis of enormous amounts of data. Any reduction in data handling offers benefits and savings in terms of time, resources, and cost. Thus a methodology that effectively reduces very large data sets to smaller, yet faithful representations of the original data sets will be an enormous asset. Further, a strategy for providing this information to plant operators, maintenance personnel, and plant management that integrates with existing plant Man-Machine Interface (MMI) systems and includes capabilities for success path monitoring of safety systems and the presentation of information generated from *smart* equipment will be necessary.

An integration of all the information to produce the "big picture" or enterprise-level view of reliability improvement in nuclear power plants is the next step. The big picture is often referred to as enterprise asset optimization, which can be simply defined as maximum asset availability and performance for the least cost. A Computerized Maintenance Management System (CMMS) is an essential part of any reliability management program since it collects data, such as labour, materials, downtime, contract costs, symptoms, failure and action information. More advanced CMMS packages based around a workflow engine are starting to be developed, and these are intended to be integrated with other plant systems such as financial, manufacturing resource planning, shop floor data collection, condition monitoring, predictive maintenance, electronic data interchange, etc. They can accumulate more data than traditional systems for reliability analysis. However, to be fully effective, reliability needs to be plugged in to the entire enterprise supply chain. These systems provide even more reliability analysis data such as production data and asset and vendor information from across the plant. Achievement of this level of integration represents a formidable challenge. To be able to perform this level of integration, it will first be necessary to develop techniques that combine "equipment health" information from individual machines into "plant health" information. While it is obviously beneficial to perform predictive maintenance on individual pieces of equipment, the ultimate goal is to develop methodologies to combine predictive maintenance information into a plant-wide system that includes the capability to assess the impact of preventive maintenance on plant profits.

Development will be necessary to establish the methodology for systematically evaluating equipment to determine how best to improve its reliability and optimize smart equipment. From such developments should come descriptions of how to apply *smart* features to a variety of equipment. At the corporate level it becomes somewhat easier also to document these benefits of integrated *smart* analysis, comparing the investments and costs incurred with the revenues that accrue from cost savings and performance improvements from plant health monitoring systems. From results of the above programmes, in part or in full, an estimate of the overall reliability benefits that could be expected for a typical new nuclear plant can be prepared.

In summary, advanced and *smart* technologies offer the tools to significantly reduce costs by streamlining design, procurement and construction phases and improve the efficiency of plant operation, all of which are necessary to lower the costs of nuclear generated electricity. Further development will be needed to achieve regulatory acceptance of *smart* technologies; for example, the signals from the "smart" systems must be correlated with reliability, and criteria must be developed for when to do maintenance and replacement.

Annex 20 describes a programme being carried out under the U.S. Department of Energy's Nuclear Energy Research Initiative to develop a set of tools and methodologies that can improve reliability and safety of nuclear power plants through the use of "smart" equipment and predictive maintenance technology.

3.2.3. Application of passive systems

The application of passive safety systems [26], i.e., those whose operation takes advantage of natural forces such as convection and gravity, is likely to be one of the most significant ways 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, holds an important key to future cost reduction, a key whose value can be verified by the use of PSA.

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’ which 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: Strategies for the Future’ [19] 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. [5]. 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 which 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 [27] [28] [29]. 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 thermohydraulic phenomena which 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 subcooled water.

Code benchmarking activities on an international level are useful to assure proper modelling of passive components and systems.

3.2.4. Re-evaluation of user design requirements with a focus on economic competitiveness

Beginning in the mid 1980s and continuing into the 1990s, user requirements documents for advanced light water reactors have been prepared. They include the EPRI ALWR Utilities Requirements and the European Utility Requirements. These documents have established sets of requirements that have been used to guide the designs of evolutionary LWRs [3]. These documents have built on the accumulated experience of plant licensing, construction, operation and maintenance, and have incorporated developments in technology and safety. They reflect lessons learned on current plants and provide the set of assumptions to be used to assess compliance with acceptance criteria.

However, user requirements can also be a cause for over-design and high capital costs. Cost benefit analysis of different design solutions has not always been carried out to determine if the requirement is in fact cost effective, or to determine whether the requirement could be met more cost effectively.

For example, user requirements documents for future plants specify plant availability factors of 87% and above, and identify design features for achieving these requirements. In fact, many well managed current plants are achieving availability factors of 87% and above. Advances which have been applied at current plants, such as high burnup fuel, better man-machine interface using computers and improved information displays, and better operator qualification and simulator training, will certainly be incorporated into new designs and will contribute to high availability figures for future plants.

Design features requested in the requirements documents for future plants for achieving the availability goals include:

- increased thermal and hydraulic design margins;
- redundancy – reducing vulnerability to single component failure, and enabling maintenance and repair during plant operation;
- diversity – reducing the sensitivity to "common-mode" failures;
- plant layout and installations that ensure accessibility for inspection, maintenance, replacement and repair;
- design for on-line testing and maintenance; and
- design for reduced in-service inspection[18].

While these features can be incorporated at the design stage to contribute to high plant availability, it is important to note that all of them, except the last two, result in added capital costs. In most designs for new nuclear power plants, margins have been introduced to enhance safety or achieve higher availability, but with added cost. The same is true for increased redundancy and diversity. A re-evaluation of requirements in these areas may well lead to major cost savings without compromising the safety or reliability of the plant.

The need for careful economic optimization can be demonstrated by considering increased design margins. Increased margins provide:

- capability to accommodate disturbances and transients without causing challenges to the plant safety, and initiation of engineered safety systems;
- an enhancement of system and component reliability, reducing the potential of exceeding specified limits which would require de-rating or shutdown;
- additional assurance that longer plant lifetime goals (e.g. 60 years) can be met.

[18] An example of a design feature for reducing in-service inspection requirements and work to be done in high-radiation areas, leading to less occupational radiation exposure for new designs, is the enhanced utilization of forgings in the pressure retaining parts of the primary system, to reduce the number and length of welds which require inspection.

While increased margin adds to the capital cost, substantial design margins can provide an option that may become attractive for the power generator once experience has been gained with the new plant: the operator can up-rate the plant power level. However, this potential advantage must be weighed against the possibility that the revenue from the additional generation may not offset the initial investment in substantial design margins.

Examples of user design requirements for which a re-evaluation may lead to cost savings are:

- requirements for measures to address corrosion of steam generator tubes that have been set for both low (hot leg) RCS temperature and for improved corrosion resistant steam generator materials, while the improved materials could in fact operate reliably at somewhat higher temperature;
- requirements for a very high number of cycles to be considered for load-following and frequency control which are sometimes much higher than could be realistically expected during the plant lifetime;
- requirements for large margins and specific additional design features imposed for the purpose of achieving high availability targets, considering that well managed existing plants have achieved high availability without such measures;
- requirements for increased safety margins for cases in which assumptions used to assess compliance with acceptance criteria are already very conservative (e.g. assumptions for reactor coolant pump flow, heat transfer surface areas, heat transfer coefficients, etc.);
- use of excessively conservative assumptions in assessing compliance with acceptance criteria. A typical example is the requirement to assume a very large number of steam generator tubes to be plugged (much higher than anticipated at end of plant life) in the analyses of plant accidents;
- prescriptive requirements for certain design features against severe accidents, without regard to the probability of certain sequences (e.g. the deterministic requirement to design the containment to withstand an in-vessel steam explosion);
- core management objectives which lead to the need to consider exceptionally high peaking factors in safety analyses which must assume mis-positioning of fuel assemblies;
- requirements for diversity of equipment which can have some negative consequences including added complexity of components and systems, and increased operator burden during maintenance. For example, the number of different spare parts needed is increased, which in turn leads to increased costs, and it also introduces risks that maintenance personnel will use a wrong component, make an erroneous adjustment or that operators will use a wrong manual or operating procedure. With an increasing number of somewhat similar though different components the personnel's familiarity with the equipment and its functions will decrease. An evaluation of such user requirements for diversity should be consistent with regulatory design requirements[19].

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[19] The IAEA document "The Safety of Nuclear Power Plants: Design"; Safety Standards Series No. NS-R-1 (1999) deals with the potential for certain common cause failures and states that the reliability of some systems can be enhanced by using the principle of diversity. It recognizes that "care should be exercised to ensure that any diversity used actually achieves the desired increase in reliability in the implemented design. For example, to reduce the potential for common cause failures the designer should examine the application of diversity for any similarity in materials, components and manufacturing processes, or subtle similarities in operating principles or common support features. If diverse components or systems are used, there should be a reasonable assurance that such additions are of overall benefit, taking into account the disadvantages such as the extra complication in operational, maintenance and test procedures or the consequent use of equipment of lower reliability".

Re-evaluation and modification of user requirements could result in more cost effective designs. This could also result in relief on Technical Specifications, or other potential benefits for the plant operator.

3.2.5. Improving the technology base for eliminating over-design

Improved safety analysis calculational tools and databases can provide more accurate predictions of plant behaviour during normal operating and accident conditions. This can facilitate more economic designs for future plants by 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. This is a broad field, and there are numerous approaches for improving calculational methodology and databases that have recently been, and are currently being developed. In general, these include 3-D kinetics codes for core physics analyses, coupled neutronic-thermalhydraulic codes for open channel designs, 3-D kinetic codes coupled with plant system codes, and 3-D Computational Fluid Dynamics (CFD) codes for analysis of passive heat transport systems. Specific examples include use of 3-D codes together with Code Scaling Application and Uncertainty (CSAU) to provide best estimate results rather than use of point reactor kinetics models with conservative approximations and boundary conditions. Another example involves sub-channel analysis of fuel rod assemblies with detailed mechanistic modelling (e.g. with two-fluid, three-field flow models, spacer grid models, cross flow models and liquid film and droplet models) to predict boiling transition and re-wetting phenomena. Other work is underway to improve the thermal-hydraulic relationships and the databases of thermo-physical properties used in the analytical codes. On the other hand, experimental data for benchmarking of these calculational methods are needed (for example, data on void generation in open channels and its reactivity feedback).

3.2.6. International consensus regarding commonly acceptable safety requirements that would facilitate development of standardized designs

There are NPP design and construction organizations in only a few countries. These organizations are becoming increasingly dependent on international trade for their commercial success, with the goal of exporting their products and services. Exports and often export credits are integral to plans for new plant construction. Yet the design and construction of each new NPP entails a complete safety and licensing review in the importing country — a process which can be time consuming and costly. More importantly, the licensing review can result in a plant design that is different from a similar version licensed in another country. Thus, it becomes difficult to deploy a standardized design internationally, without producing different variants for each country. This significantly adds to nuclear plant costs. As electricity markets become more competitive, the plant costs associated with country by country licensing need to be minimized. In effect, the future of nuclear power as a viable generating technology is contingent on establishing means to facilitate the building of new NPPs when and where they are needed with a minimum of delay.

The cost to develop a new nuclear plant design and bring it to market is very substantial. Designers focus on optimizing safety and economics and on streamlining plant construction and operation. To truly optimize safety and economics, there is a need to assure that new reactor designs can be marketed in as many countries as possible — without requiring re-design every time that a design is introduced into a new country. Design organizations need to be assured that their design costs can be recouped from sales in as many countries as possible.

Being able to design to a single set of safety requirements would provide a substantial incentive to develop new designs. To support this objective, it would be highly desirable to achieve international consensus regarding commonly acceptable safety requirements that would facilitate development of standardized designs.

Section 3.1 notes that standardization can be a powerful tool in reducing costs. Standardization allows first of a kind design costs to be spread over a large number of units. It provides economies-of-series production, by allowing equipment to be manufactured in bulk quantities and plants to be constructed in an assembly-line fashion — with the learning curve from the first units serving to reduce the costs of all of the follow-on units of the same design. It reduces the costs of storing spare parts and for training maintenance and operating personnel. International consensus regarding commonly acceptable safety requirements would facilitate development of standardized designs and allow them to be used in much larger markets – reducing costs for owners, investors and safety authorities.

The process should build on collective international experience in nuclear plant design and safety regulation to produce a set of internationally commonly acceptable standards that achieve a very high safety level while at the same time avoiding requirements that complicate the design and add to plant cost without providing significant safety benefits. As discussed in Section 3.2.1, applying PSA to safety and design requirements could lead to substantial design simplification for new nuclear plants, while still maintaining high levels of safety. Moving to a uniform (but smaller) set of less prescriptive safety requirements that could be accepted in any country would greatly facilitate nuclear power as an economically competitive energy source in the global marketplace.

The IAEA has, over the years, issued a substantial number of international Safety Standards (Requirements and Guides) that are useful to all countries utilizing (or even considering) nuclear energy plants. Although individual countries are free to use IAEA Safety Standards as they see fit, the Safety Standards are available to every country and could serve as reference material for developing further international consensus on all safety related aspects.

4. IMPLICATIONS FOR THE NUCLEAR COMMUNITY — LEARNING NEW WAYS AND FINDING A NEW BALANCE

There is a shared interest in the nuclear community for nuclear power to continue as a viable participant in future electricity markets. This implies an industry that is forward looking and change-oriented at all levels, including investors, managers, designers and regulators, and a focus on plants that generate power efficiently, safely, profitably and at competitive costs. It also implies the need for all members of the nuclear community to re-examine their assumptions, goals and practices, to find new approaches, by working together, that are consistent with today's and tomorrow's commercial realities.

Design organizations are challenged to develop advanced reactors with

- considerably lower capital costs and shorter construction times;
- simplified designs which achieve high safety levels in the most cost effective manner;
- sizes (including small and medium sizes with load following capability) appropriate to grid capacity and owner investment capability;

— high levels of standardization and modularization incorporating the latest technological advances.

Regulatory agencies are challenged to

— move to more risk-informed safety requirements for new plants;
— establish design certification procedures which don't take too long, and which are sufficiently flexible to accommodate
  — incorporation of advances in technology,
  — design changes which maintain adequate safety level at reduced cost, and
  — a variety of users needs (e.g. needs for a variety of plant sizes); and
— establish international consensus regarding commonly acceptable safety requirements that would facilitate development of standardized designs, so that such designs could be used in several countries.

Power generating organizations are challenged to set user requirements for new designs that result in the most cost effective solutions while meeting safety requirements and which are profitable under changing market conditions.

Appendix

TECHNOLOGIES FOR EXTENDING PLANT LIFETIME, IMPROVING AVAILABILITY AND REDUCING OPERATION AND MAINTENANCE COSTS

With deregulation and privatization of electricity markets, nuclear plant operators are experiencing an operating environment with increased competition from other suppliers of electricity. This competitive environment has significant implications for plant operations, including efficient use of all resources; more effective management of plant activities, such as outages and maintenance; and sharing of resources, facilities and services among power generators. The overall result is a significant reduction in operation and maintenance costs.

Important means of improving economics involve technologies for plant life extension, and for increasing plant availability. Both require attention to both the nuclear island and the balance of plant. Nuclear power plants worldwide are improving their energy availability factors. The world average has increased from below 70 percent in 1983 to 82 percent in 2000[20], with some power generators achieving significantly higher values. This is being achieved through integrated programmes including personnel training, quality assurance, improved maintenance planning, longer fuel cycles, as well as technological advances in plant components and systems, and in component inspection, maintenance, repair and replacement techniques.

International co-operation is playing a key role in this success. The various programmes of the World Association of Nuclear Operators (WANO) to exchange information and encourage communication of experience, programmes of the OECD/NEA and the European Commission to improve component inspection techniques, and the activities of the IAEA including projects in nuclear power plant performance assessment and feedback, effective quality management, and information exchange meetings on technology advances [30] [31] [32], are important examples of international co-operation to improve plant performance.

Technologies for extending the lifetime of current plants

The economic life of a plant is defined by the market, and the cost of continued operation of the plant may not coincide with the period of the plant license. Different countries take different approaches in setting the duration of the operating license for a nuclear plant. In some countries, there is a "licensed life" with possible consideration of life extension, while in others there is a "periodic safety assessment" to approve the plant for a further fixed period of operation. Political decisions to end the operation of a plant before it reaches its technical or economic lifetime also occur. Provided the economics of the plant are favourable, there is a considerable incentive to seek life extension.

Life extension up to a total of 60 years is being pursued in several countries. In the USA, the Nuclear Regulatory Commission has already issued twenty-year extensions of the operating licenses for several nuclear power plants. It is anticipated that most nuclear plants in the USA will seek similar extensions of their operating licenses over the next several years. The

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

incentive for seeking license extensions well in advance of the expiration of the initial 40-year license, is quite simple to understand. If the plant owner knows he will be allowed to continue operating for a longer period of time, there is less risk in making capital investments needed for future operations.

Life extension requires qualification of components and structures related to safety performance supported by fatigue analysis and ageing effects analysis. Components addressed include the reactor pressure vessels of LWRs or the pressure tubes of HWRs, reactor internals, steam generators, pumps and valves. Life extension typically involves at least particular component replacement. Because the reactor pressure vessel cannot easily be replaced, vessel annealing has been an important approach for alleviating vessel embrittlement with age. HWR pressure tubes have been replaced in 5 units, and experience now predicts that this can be achieved with a 9 to 12 month shutdown after about 30 years of operation.

The ability to qualify systems, components and structures for life extension is facilitated by life management programmes that monitor and establish their conditions. Key elements of these programmes include:

— periodic inspections of major components such as the LWR reactor vessel, HWR pressure tubes, primary system piping and steam generator tubing;
— effective maintenance;
— development of inspection and repair technologies, and the acquisition of material data and operational data related to ageing.

Materials technology programmes have been established by nuclear plant operators and government organizations focused on understanding and managing materials condition and performance issues with the goals of determining and increasing component residual service life. Specific focus has been on components such as pressure vessels and internals, reactor pressure tubes, primary system piping and steam generator tubing.

Each of the principal light water and heavy water reactor types has experienced material-related problems during the service life of the initial versions. These have been overcome in various ways including replacement with more resistant materials and changes in the chemistry of the water environment or by design modifications based on full scale model testing.

BWR reactors, for example, have experienced cracking of reactor internals (e.g. the core shroud) made of type 304 stainless steel due to inter-granular stress corrosion cracking. Inspection, repair and replacement techniques have been developed using extensive laboratory, and in-plant data as well as full-scale mock-up test facilities to detect and correct the damage.

PWRs have suffered from cracking of vessel head penetrations, core barrel and bottom mounted instrumentation adapters. Reactor vessel head replacement has been needed at several PWRs because the nickel-based Alloy 600 sleeve material was susceptible to stress corrosion cracking. The replacement heads include improved materials for penetrations (e.g. Alloy 690) and integrated forged head designs.

HWRs have experienced problems related to pressure tube and boiler tubes. The condition of the initial pressure tube alloy (Zircaloy-2) deteriorated from the failure of tube supports to prevent thermal gradients from developing in the tube and causing hydride formation and cracking. This problem necessitated pressure tube replacement in 5 units with improved designs of spacer tube supports. The replacement alloy was Zr-2.5Nb, which picks up significantly less corrosion hydrogen than Zircaloy-2 during service.

Although significant progress has been made in understanding irradiation and thermal degradation of LWR reactor vessel steels, some aspects are still not fully understood. In particular, further work is essential on the qualification of remedial measures such as annealing and repairs. The international efforts of the IAEA Working Group on Nuclear Plant Life Management and the OECD Nuclear Energy Agency (NEA) Principal Working Group 3 (PWG3) provide national contacts between institutions working in this field. The European Network AMES programme (Ageing Materials Evaluation and Studies) was initiated in 1993 including material studies to improve the understanding of the effects of irradiation damage, ageing and annealing, micro-structural model development and studies on irradiation and thermal degradation of materials for new reactors.

The pressure tubes of HWRs have and continue to receive considerable study to determine life limits before fullscale replacement. Deformation by irradiation enhanced creep and growth of the zirconium alloy tubes limits the life to about 30 years although development results predict a steady increase.

The dominant cause of damage to pressurized water reactor steam generators has been corrosion of the tubing and support structures. The alloy, Inconel 600, has suffered extensive corrosion failures in many plants, sometimes requiring steam generator replacement. Large efforts in several countries have been and are being carried out to control and improve the service environment to extend the service life of steam generator tubes. New alloys (e.g. alloy I800 and Inconel 690) have been shown to have superior corrosion resistance compared to Inconel 600 and are now favoured for new and replacement PWR steam generators.

With regard to in-service inspection of primary circuit components (especially the reactor vessel, primary piping and steam generator tubes), the EC and OECD/NEA have conducted major international efforts for more than 20 years to improve the capability and reliability of non-destructive evaluation methods. Improvements have been introduced through procedures specifically adapted to the defects to be detected.

A major international effort to improve the assessment of the capability and reliability of inspection techniques and procedures for non-destructive evaluation of structural components has been carried out since 1974 in the Programme for the Inspection of Steel Components (PISC) by the European Commission in co-operation with OECD/Nuclear Energy Agency. Inspection methods for pressure vessels, dissimilar metal welds, primary piping welds, and steam generator tubes have been evaluated. This work culminated in the publication of the first edition of the European Methodology for Inspection Qualification in 1995, and the second in 1997.

For HWRs, specific inspection techniques have been developed to detect flaws in the relatively thin-walled zirconium alloy pressure tube tubing and primary circuit tubing most of which is also small diameter (<100mm dia). For zirconium tubing the inspections can use both ultrasonic and eddy current techniques to increase overall the detection sensitivity to small flaws.

Technologies for improving performance of current plants

Considerable improvement in outage time and generation costs can still be achieved through technical and administrative measures. Examples of technologies for improved performance include high burnup fuel (which supports longer cycle length), power up-rating to achieve higher output, computer-aided systems to provide early indication of sensor or component degradation, and simpler systems for PWRs and HWRs for control of hydrogen during accident conditions (systems that require considerably less testing and maintenance and thereby reduce outage duration).

Power up-rating often involves replacement of some heat transfer and power conversion equipment, as well as analyses required to support the license to operate at higher power levels. Steam turbine efficiency has increased over the past 20 years, and remodelling of turbines can result in an increase of power output without any upgrading of the reactor. Many operators have made technical improvements to take advantage of power up-rating. In Sweden, for example, the nuclear industry has added approximately 600 MW(e) of capacity by improving its existing stations.

Improved performance at current plants is also supported by implementation of activities to analyse information from operation of components and systems to understand the causes of unavailability, and to improve work processes during maintenance.

Procedures for planning and carrying out maintenance influence reliability and availability. Good planning and organization of work during planned outages can strongly contribute to shortening outage duration and thereby contribute to availability improvements. For example, formation of special outage management teams with a focus on close communication influences the efficiency of the execution of work during an outage. Efficient outage management also deals with logistics support including requirements for special tools and availability of spare parts as these factors influence outage duration.

New I&C and control room technologies which can contribute to improved plant performance are being backfitted into operating plants. These include:

- digital instrumentation and control, including self-diagnostic systems; and
- control room and man-machine interface improvements with due consideration of human factors engineering.

Changes in regulatory policy can also facilitate improvements in plant reliability and availability. As an example, The U.S. Nuclear Regulatory Commission's new Risk-Informed Performance-Based regulation policy is expected to contribute in this area because it is a goal-oriented, rather that a means-oriented system focusing on what licensees must achieve rather than on what they must do. Simplification of technical specifications is an application that should reduce the number and length of plant outages, as well as the number of plant personnel required to implement the ‘tech spec’ requirements. For example, simplifications in surveillance and maintenance activities for emergency diesel generators are expected to result from the goal oriented practices of this new policy. Another benefit from this new policy may be the ability to reclassify many of the systems, structures, and components in the nuclear plants, so that they are not required to fully satisfy all of the special conditions normally imposed upon safety-grade equipment. Instead, many of them could be purchased and maintained to commercial quality standards, in the future.

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ANNEXES

Annex 1

Turkey's recent decision regarding the Akkuyu NNP

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Abstract. The Turkish Government postponed the Akkuyu NPP project on 25th of July 2000. This was the 4th unsuccessful attempt of the country to build a nuclear power plant since 1965. The estimated burden of the foreign credit on the economic program to stabilize the economy and drop the inflation rate was the main reason for this decision. The Government stated that the postponement of the Akkuyu NPP project did not mean that Turkey would avoid usage of nuclear energy in the future. The Cabinet's announcement also states the need of contributing to the technological improvements of new generation nuclear power plants.

1. HISTORY

Studies to build a nuclear power plant in Turkey were started in 1965. Later, between 1967 and 1970, a feasibility study was made by a foreign consultant company to build a 300-400 MW NPP. The NPP would have been in operation in 1977. Unfortunately, because of the problems relating to the site selection and other issues the project could not come into life.

In 1973, the Turkish Electricity Authority (TEK) decided to build an 80 MWe prototype plant. However, in 1974 the project was cancelled because this project could delay the construction of a greater capacity nuclear power plant. Instead of this prototype plant, TEK decided to build a 600 MWe NPP in southern Turkey.

Site selection studies were undertaken in 1974 and 1975 and the Gülnar-Akkuyu location was found suitable for the construction of the first NPP. In 1976, the site license for Akkuyu was granted by the Atomic Energy Commission. In 1977, the bid was prepared and ASEA-ATOM and STAL-LAVAL companies awarded as the best bidders. Contract negotiations continued until 1980. However, in September 1980 due to the Swedish Government's decision to withdraw the loan guarantee the project was cancelled.

The third attempt was made in 1980. Three companies were selected to build four nuclear power plants (1 unit CANDU (AECL) and 1 unit PWR (KWU) in Akkuyu and 2 units BWR (GE) in Sinop). Due to Turkey's suggestion of the BOT model, KWU resigned from the bid. Although AECL accepted the BOT model, it insisted on the governmental guarantee for the BOT credit. The Turkish government refused to give the guarantee, thus the project was cancelled.

In 1992, the Ministry of Energy and Natural Resources stated in a report submitted to the Government that without the installation of new energy resources before 2010, the country would face an energy crisis, suggesting that nuclear energy generation should be considered as an option.

In 1993, The High Council of Science and Technology established the nuclear electricity generation as the 3rd highest priority project of the country. In view of this decision, the Turkish Electricity Generation and Transmission Company (TEA[a]) included the NPP project in its 1993 investment program. In 1995, TEA[a] selected the Korean KAERI as the consultant for the preparation of the bid specifications. The bid was started in 1996. Three companies made proposals in 1997: AECL, NPI and Westinghouse. After a series of delays the Government decided to postpone the project in July 2000.

2. THE NEED FOR NUCLEAR ENERGY

During the period of 1996-2000 the primary energy consumption in Turkey increased by 4.5 % per year and reached to 78.8 M-toe by the year 2000. During the same period, the electricity demand increased about 8.2 % per year, and is expected to reach about 127 TWh at the end of this period. The primary energy consumption per capita is about 1.2 toe and the electricity consumption per capita is about 1,900 kWh as of year 2000. Electricity production has been rising steadily; it grew from 111 TWh to 116 TWh during the period of 1998-1999. The installed capacity, during the same period, increased from 23,352 MW_e to 26,117 MW_e. The installed capacity and electricity production in 2000 is expected to be about 27,400 MW_e and 124 TWh, respectively. It is expected to have an annual demand rate of about 8 % - 10 % till 2010. The projection for electricity consumption reveals that about 290 TWh will be consumed by the year of 2010. Thus nuclear energy is most important alternative to fossil resources when diversity and energy supply security are taken into consideration. Today, the electricity generation composition is about; 70 % thermal (coal, gas, oil, geothermal) and 30 % hydro.

It is well known that fossil fuel utilization is dominant over other energy types and its share in the global electricity generation is around 65 %. The main draw back of fossil fuel utilization is the environmental pollution and especially the CO_2 emission. Today the total CO_2 emission reached to about 22,000 M-ton (about 3800 kg/capita) and the share of power generation is about 8,000 M-ton. The nuclear energy is an important option and alternative to fossil fuels provided that the economical aspect of a NPP is improved so as to become more competitive in deregulated market and the problem of public acceptance is solved. The economical aspect of the nuclear power is highly significant for industrializing countries like Turkey since capital cost share of nuclear electricity generation is about 60% - 70%, contrary to the figure that fossil fuel plants have, i.e. 20% - 40%. This fact endangers the NPP projects in developing countries since external credit is unavoidable in those countries and credit guarantees and reimbursement of credit plays the central role in making decisions on NPP projects impossible or hardly possible in those developing countries.

3. POSTPONEMENT OF THE AKKUYU PROJECT

In spite of the fact that nuclear energy contribution was planned to be 9,000 MW_e by the year 2020 (a 9 % share to total generation) and there was a strong intention of the Government to install our fist NPP in Akkuyu, the Government decided to postpone the Akkuyu NPP project, following the meeting of the Cabinet held on 25th July, 2000. The Government's statement on this decision made it clear that the reasons were not related to safety issues. Since Turkey needed to concentrate on a program of economic stability aiming to reduce inflation rates to reasonable figures, the government could not afford the estimated three to four billion US dollars needed for construction of the country's first nuclear power plant. The suggestions that the project had been dropped because of fears of earthquakes in the region are "unfounded".

Akkuyu is a remote site in the least seismic area of the country and the continuous re-evaluation of the site has been ongoing, since the mid 1970s using the latest analytical techniques. Moreover, the argument about the NPP's negative effect on Turkish tourism is neglected in view of the good example of France. The Government also stated that it was more preferable to build natural gas power plants in the short term, like other OECD countries. However, in the long run, i.e. 15-20 years period, if natural gas becomes scarce and less economical then it will be better to reconsider the nuclear power option. According to the Turkish Government, in order to be prepared for a natural gas crisis construction of large numbers of NPPs will be required. Since the country's resources are limited, it is not possible to achieve such projects without external loans and such huge external loans might endanger economic programs. Therefore, It is better to continue hydro and natural gas projects and wait for the decrease in NPP costs and increase in their lifetimes. During this period, there may rise an opportunity to utilize our thorium reserves as a nuclear fuel. The Government also stated that it is not planned to cancel our plans to build NPPs.

4. FUTURE PLANS

The Turkish Electricity Generation and Transmission Corporation has been studying a new energy generation plan since the postponement of the Akkuyu project. It is also declared by the Government that the postponement of the Akkuyu NPP project does not mean that Turkey will avoid using nuclear energy in the future. The Cabinet's announcement also states the need of contributing to the technological improvements of new generation nuclear power plants.

Since the future nuclear power program of Turkey is to be dependent on nuclear policy, the Turkish Atomic Energy Authority (TAEA) recently initiated a project to revise the nuclear policy of the country. This project will include the application sectors of nuclear energy, including nuclear power, and programs associated to each sector. One of the sectors that should be considered is "*Research and Development*" which also includes innovative designs and small and medium sized reactors (SMRs). As mentioned in the IAEA documents SMRs have the following technical features:

- Lower absolute capital cost with a smaller financial burden;
- Co-generation of electricity and heat for district heating;
- Distribution of economic risks through several smaller plants;
- Better controlled construction schedule (2-4 years) due to less on-site work and smaller size of components;
- Earlier introduction of nuclear power by use of SMRs help earlier returns of investment and will serve for environmental protection against fossil fueled plants;
- Better fit to smaller and weaker grids;
- Fit to low load growth rate situations;
- Better past performance records than larger plants (148 SMRs in operation and 12 SMRs under construction);
- Earlier introduction of nuclear power with potential for short term technology transfer;
- Plant life extension can be made possible by allowing construction of modular reactor vessel for replacement;
- Decommissioning cost for smaller components would be affordable and even the reactor vessel (for reactors up to 300 MWe) could be transported to some central site for dismantling.

Cooperation with international/national groups on theoretical and experimental projects concerning SMRs and innovative technologies would lead to an increase of staff capabilities and experience on nuclear technology. To achieve this goal, TAEA decided to participate in the "International Project on Innovative Nuclear Reactor Technologies and Fuel Cycles", a new project of the IAEA.

Annex 2

Building a new nuclear power plant in Finland? Studies performed

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Abstract. The electricity consumption per capita is high in Finland due to the country's industrial structure and to the climatic conditions. Industry consumes 55% of the electricity in Finland. The demand of electricity is expected to grow at a rate of 1.5% a year until 2010, and further at a yearly rate of 1% until 2015. This will require 3800 MW of new generating capacity by 2015. A recent study indicates that in base-load power production in Finland the generating costs of a nuclear plant are the lowest in comparison with generation using coal, natural gas or peat. The difference to coal would be 9%, to gas 18% and to peat 40%. The target for Finland to reduce greenhouse gas emissions under the EU burden sharing is 0%. In comparison with business as usual scenarios, however, the reduction need is of the magnitude of 20%, one of the hardest in the EU. Finland already has taken into use the methods, which now are considered essential within the EU for reducing the CO_2 releases. Teollisuuden Voima Oy (TVO) submitted on 15 November 2000 to the Council of State an application for a decision in principle concerning the construction of additional nuclear capacity. The submission of the application is reasoned by the shareholders' need for additional electricity. Furthermore, nuclear power, together with renewable energy sources, makes it possible to comply with the Kyoto protocol commitments. The actual investment decision can be made first after a positive decision in principle has been received from the Council of State and the Parliament. The submission of the application was preceded by a number of studies, the contents of which are summarised.

1. WHY NEW NUCLEAR POWER IN FINLAND

The electricity consumption per capita is high in Finland due to the country's industrial structure and to the climatic conditions. Industry consumes 55% of the electricity in Finland. The high share of industrial consumption maintains a high demand for the base-load power. The consumption of electricity has been increasing continuously in Finland with short stagnations during the years after the oil crisis and during the heavy economical depression in early 1990s. During the last decade the average yearly growth of electricity consumption was 2.2%. The demand of electricity is expected to grow at a rate of 1.5% a year until 2010, and further at a yearly rate of 1% until 2015. This will require 3800 MW of new generating capacity by 2015.

In addition to fuel imports, Finland has continuously been a net importer of electricity from the Nordic and Russian markets. The share of imported electricity was 14.3% in 1999. Finland is part of the liberalised Nordic electricity market. Electricity supply in the area of Nordic countries is dominantly based on hydropower. The availability and the price of electricity on this market are highly dependent on the amount of hydropower. The difference between a rainy year and a dry year can be as high as 74 TWh, which is almost the yearly electricity consumption in Finland, and means an uncertainty in the security of the Finnish electricity supply. Total generation capacity in use in the Nordic market is at the moment abundant, but demand is estimated to outgrow supply around 2005 under normal water conditions. The dependence of natural gas, now 10% of primary supply, has been growing and the growth continues. The only supply of natural gas to Finland comes via a pipeline from Russia.

A recent study by Lappeenranta University of Technology indicates that in baseload power production in Finland the generating costs of a nuclear plant are the lowest in comparison with generation using coal, natural gas or peat. The difference to coal would be 9%, to gas 18% and to peat 40%. The comparison includes capital, fuel, operating, and waste management costs. In the financial analysis a real interest rate of 5% per annum and the plant full load utilization time of 8000 hours per year were used.

The target for Finland to reduce greenhouse gas emissions under the EU burden sharing is 0%. In comparison with business as usual scenarios, however, the reduction need is of the magnitude of 20%, one of the hardest in the EU. With the extensive use of combined heat and power and the large share of the renewable energy sources in the electricity production, Finland already has taken into use the methods, which now are considered essential within the EU for reducing the CO_2 releases.

The former Government's energy strategy, which was approved by the Parliament in 1997, puts the first priority on increasing the use of natural gas but also makes clear the necessity for keeping the nuclear power option open, especially for the case where the supply of gas cannot be guaranteed. The policy of the ruling Finnish Government also keeps all alternatives open for the future electricity production putting the priority on methods, which help in limiting the pollution of the atmosphere. No options that are technically, economically or environmentally feasible should be excluded.

2. APPLICATION FOR A DECISION IN PRINCIPLE

According to the Nuclear Energy Act, a company considering a nuclear plant project must apply for a decision in principle from the Government on beforehand. The Government decides whether the project is in accordance with the overall good of the society. If the decision is positive, it needs ratification by the Parliament. Before the Government decision, various interested parties are heard, including the municipality of location and the Radiation and Nuclear Safety Authority, STUK. The entire process takes 1-2 years. Only after the ratification of the decision in principle the company can proceed to apply for the construction permit from the Government.

Teollisuuden Voima Oy (TVO) submitted on 15 November 2000 to the Council of State an application for a decision in principle concerning the construction of additional nuclear capacity. The submission of the application is reasoned by the shareholders' need for additional electricity. Furthermore, nuclear power, together with renewable energy sources, makes it possible to comply with the Kyoto protocol commitments.

TVO applies for a decision in principle for a nuclear power plant unit, which is either of BWR or PWR type. The electric output of the unit is, depending on the plant type, 1000-1600 MW. The plant unit will be located either at the Loviisa or Olkiluoto nuclear power plant site. The cost estimate for the new unit is FIM 10-15 billion (EUR 1.7-2.5 billion), depending on the plant size. TVO will finance the project. The actual investment decision can be made first after a positive decision in principle has been received from the Council of State and the Parliament.

3. Studies on new nuclear power plant

The submission of the application was preceded by a number of studies. In early 1990s bids for a new nuclear power plant were asked and received, but the project was rejected by the Finnish parliament in 1993. New studies were initiated in mid 1990s, and an approach developed as a preparation programme was adopted in late 1990s. Studies were made both by Imatran Voima Oy (presently Fortum Oyj) and TVO. In 1999 it was agreed that TVO would be responsible for the preparations and implementation of a possible project to build the new power plant.

When considering building a new nuclear power plant, an indispensable issue is safety. Besides safety, special attention must be paid to economy in the plant design. The new plant designs have been simplified and the time needed for construction work has been reduced. Consequently, the following criteria were set for the suitability studies:

- The new NPP unit shall be
    - economically competitive
    - safe and licensable
    - on line in due time with respect to the increasing electricity demand
    - compatible with the grid;
- Possible additional units shall be taken into account in location and implementation;
- Infrastructure at the site shall be utilised as much as possible;
- Failure risk of the project shall be as low as possible.

These criteria reflect the specific situation in Finland. A new NPP unit would be built at either of the present sites, Loviisa or Olkiluoto, where even additional units can be located. Utilisation of the existing infrastructure and the same procedures and facilities that are used for nuclear waste management of the present plants will reduce the investment and production costs of the new unit. Only one unit will be considered for the time being, although the increasing electricity demand shows that the possibility for another unit must be maintained.

4. PREPARATION PROGRAMME

The purchase process of a new nuclear power plant consists of activities that can be grouped into the following subsequent phases, where the first two phases are related to the preparation of plant purchase and building:

- preparedness phase
- decision-in-principle phase
- purchase phase
- implementation phase.

The preparedness phase started in 1998 with the main goals to increase the readiness to proceed in the purchase of the next nuclear power plant and to facilitate a prompt start of the next phase, if so decided. The studies made by TVO were first concentrated in BWRs, but since the Fortum-TVO agreement in 1999 also PWRs were included.

The preparedness phase consisted of several projects. In the following list the final outcome is mentioned.

- Environmental Impact Assessment (EIA) procedure: EIA reports for Loviisa and Olkiluoto (see *Appendix 1*);
- Site investigations: site reports for Loviisa and Olkiluoto;
- General design criteria: general design criteria report;

- Feasibility studies for various plant concepts: ABWR, BWR 90+, EP1000/AP1000, EPR, SWR 1000, VVER 91/99;
- Preliminary project implementation plans: reports on project management, schedules etc.;
- Licensing studies: application for a decision in principle;
- Communication and information studies;
- Evaluation of investment and electricity production costs;
- Preliminary financing plans.

The next phase of the preparation programme, which is related to the decision-in-principle procedure, has now been commenced with the submission of the application. Several of the above-mentioned projects contributed to the preparation of the application. E.g., the EIA reports and the plant descriptions compiled for the feasibility studies are part of the decision-making material required in this procedure. In addition to the activities that are directly connected with the licensing process, bid specifications are prepared and contacts kept with the vendors. In this way, readiness is developed to submit bid invitations as soon as needed after a positive decision by the Parliament.

5. BASIC DESIGN REQUIREMENTS

The technical studies carried out within the preparation programme covered a wide range of activities with the main objective to assure that the alternatives considered fulfil the relevant feasibility criteria.

One of the first steps was to fix the basic design requirements. These will consist of three principal parts:

- general requirements
- nuclear island requirements
- power generation plant requirements.

The first part will present the most important requirements related to main design characteristics, safety and licensing, operational targets and site conditions. The hierarchy of licensing requirements (see *Appendix 2*) will be one of the issues discussed here. Concerning site conditions, the requirements specified in the existing safety reports of Loviisa and Olkiluoto NPPs must be fulfilled. In addition, the information included in the environmental impact assessment procedures for Loviisa 3 and Olkiluoto 3 must be considered.

The general requirements were compiled in a document, the contents of which are presented in *Appendix 3*. The most crucial requirements include the following:

- Plant type: BWR or PWR
- Plant output: from about 1000 MWe to about 1600 MWe
- Site: Loviisa or Olkiluoto
- Cooling: sea water
- Technical lifetime: 30 years for major components, 60 years for components that are difficult to replace.

The nuclear island requirements will be specified applying the approach of the EUR document (*Appendix 4*). While EUR defines the requirements for a European standard NPP, the requirements for a Finnish NPP must be adapted to the local conditions. This implies several modifications to the EUR text. In addition, the lessons learned during the EUR Volume 3 work must be taken into account. The dialogue that has taken place between the NPP vendors and utilities during the Volume 3 projects facilitates this. The application of EUR model means that the contents of nuclear island requirements more or less follow the contents of EUR Volume 2. So far, no decisions have been made concerning the power generation plant requirements.

6. FEASIBILITY STUDIES

Another set of technical studies concentrated in the feasibility of the NPP concepts concerning mainly safety and licensing, basic design requirements, plant location and project implementation. In this connection discussions were held with the Finnish nuclear regulatory body, STUK. The following parts were included in the studies:

- Comparison against regulatory requirements: most important YVL guides;
- Comparison against general technical requirements: "Selected preliminary general technical requirements for studying the feasibility of a possible new nuclear power plant unit to Finland";
- Deviation report: findings of above steps, proposal for solutions;
- Discussions with STUK: minutes of the meetings;
- Site investigation: assessment of alternative locations;
- Implementation studies: scope limits, construction methods, main schedule, design & project management system;
- Economic studies.

One of the outcomes of the feasibility studies was the general description of plant design that was prepared for each concept.

7. REDUCTION OF CAPITAL COSTS

The factors affecting NPP capital costs are discussed in the OECD NEA Report "Reduction of capital costs of nuclear power plants". In the following an approach is made to assess these factors against the studies made by TVO.

Increased plant size

An electrical output between about 1000 to 1600 MW is in accordance with this recommendation. In addition, the smaller sizes belong to concepts with passive or innovative features, which inherently should have lower specific costs. Upper limit for the plant size is set by the capabilities of the external grid.

Improved construction methods and reduced construction schedule

In the feasibility studies performed one of the focal issues has been the use of advanced construction methods, including e.g. modularisation and parallel construction. These methods contribute to reducing construction schedule. A construction time of 48 months from first concrete to power operation is required. It is obvious that the time schedule is impacted by the limitations in the procurement of heavy components, especially the reactor pressure vessel.

Design improvement

The plant concepts that have been subject to feasibility studies represent next generation reactors. In their design process, systematic studies on structural and functional design requirements have been done. Experiences from operating power plants have been taken into account. These were also considered by TVO when setting the basic design requirements. The examined plant concepts try to combine the best features of proven technology with some innovative features, such as the use of passive systems.

Improved procurement, organisation and contractual aspects

These issues have been considered only preliminarily. However, in the continued studies emphasis is put on a contracting strategy that would optimise using the expertise of all parties.

Standardisation and construction in series

A new NPP unit in Finland will be a single case, and therefore, standardisation and construction in series is not possible. On the other hand, the application of EUR document provides means for vendors to design a plant that is suitable for other utilities and sites.

Multiple unit construction

Only the present sites, Loviisa or Olkiluoto, are considered. This is parallel to multiple unit construction, because the existing infrastructure and the same procedures and facilities that are used for nuclear waste management of the present plants are utilised.

Regulation and policy measures

Regulation aspects have been taken into account from the very beginning, as is obvious from the close contacts to the regulatory body. Regarding policy measures, Finnish participation in the EUR organisation has facilitated mutual cooperation between TVO and other European utilities as well as NPP vendors.

8. STUDIES ON PRODUCTION COSTS

The operating record of Finland's four nuclear power plant units is good, and the electricity has been produced at a competitive price. The long-term stability of the electricity price due to low fuel cost is seen as a vital advantage of the nuclear electricity.

A study performed in the Lappeenranta University of Technology [1] *(Attachment A)* indicates that in base-load power production in Finland the generating costs of a nuclear plant are the lowest in comparison with generation using coal, natural gas or peat. The nuclear electricity would cost 22.3 EUR/MWh, having the margins of 2 EUR/MWh and 4 EUR/MWh compared to coal- and gas-based electricity, respectively. The sensitivity analysis reveals that the nuclear option is rather insensitive to the changes of the input data, whereas the gas alternative involves a considerable risk as a consequence of increasing gas price. The comparison includes capital, fuel, operating, and waste management costs. Taking into account the external environmental costs in the calculation would further favour the nuclear alternative.

Of the four alternatives under consideration, the nuclear option is the only one, which does not generate carbon dioxide emissions to the atmosphere. A new 1250 MW nuclear unit with 10 TWh annual production would save 8.3 million tons carbon dioxide emissions annually, if the reference is the coal-fired condensing power plant. The nuclear choice would make a major contribution for achieving in 2010 the greenhouse gas emission level in accordance with the Kyoto protocol.

From the national point of view – both in terms of economy and in terms of the Finnish compliance with its Kyoto protocol commitments on greenhouse gas emissions reductions – the nuclear choice is by far the best alternative for new base-load power capacity.

9. ELECTRICITY MARKET

In a report prepared by The Finnish Energy Industries Federation Finergy [2] *(Attachment B)* the developments in the supply and demand of electricity in Finland and Nordic countries are examined up to year 2015.

The consumption of electricity in Finland is estimated to grow from the current almost 80 TWh per year to almost 100 TWh by the year 2015. The consumption prognosis, indicating an annual growth of 1.3%, has been drawn up on the assumption that economically viable electricity conservation options have been implemented and that wherever economically possible, the use of electricity has been intensified. The need for increased production capacity is some 3800 MW in the next 15 years. Even old abolished coal fired capacity must be replaced.

Imports of electricity will not solve Finland's long-term electricity needs. Finland needs to prepare for a situation where even Sweden and Norway have to import an increasing proportion of their electricity. It is also likely that as the European electricity market opens up, the flows of electricity will start to run from the north to the south rather than from the south to the north. The proportion of imported electricity of all electricity consumption in Finland is already so high that the energy industry cannot recommend increasing the imports further.

Increasing the proportion of fossil fuels in Finnish power generation is not a generally desired option. On the other hand, the use of new, renewable sources of energy will only solve a part of Finland's need for additional electricity generation capacity. In Finland and other Nordic and EU countries, it is difficult to fulfil the obligations stated in the Kyoto Protocol on climatic issues. The electricity consumption in the Nordic countries will grow from 380 billion kWh in 1999 to some 420 billion kWh by the year 2015. The increase of some 45 billion kWh means the need for additional capacity of 7000 - 8000 MW during the next 15 years in the Nordic countries.

Alongside renewable energy sources, extensive input in research and development must be made as well as the planning of time-consuming and low-emission technologies and systems pursued. Despite this, it is difficult to see how this objective could be fulfilled without current and new nuclear power. Building additional nuclear power in Finland can be seen to be in line with the official energy policy.

10. PROJECT FINANCING

The cost estimate for the new unit is FIM 10-15 billion (EUR 1.7-2.5 billion), depending on the plant size. The estimate includes interests during construction. Based on the preliminary construction schedule, the financing of the principal investment is scheduled during a period of about five years.

The project will be financed by TVO. The great majority of the costs would be financed by loans from financial institutions, special credit institutions and capital markets. In addition, the possibility of financing from plant supplier would be exploited. New equity from shareholders would be needed only to limited extent. The financing will be arranged in two stages, separately for construction phase and operating phase, taking into account the special features for both phases. The external financing is planned to be paid back in about 30 years.

The loans of TVO are estimated to rise from the present FIM 2 billion to about FIM 12-17 billion. A large share of dept financing is possible for two reasons:

- the production costs of nuclear power are predictable and stable and
- the users of the electricity will commit themselves to buy the production for the entire lifetime of the plant.

11. BENCHMARKING: IAEA APPROACH

An IAEA approach is presented in TECDOC-1123 "Strategies for competitive nuclear power plants". *Appendix 5* lists the strategies with related techniques that are given for new plants or projects that are being considered.

REFERENCES

[1] TARJANNE, R., RISSANEN, S., "Nuclear power least-cost option for base-load electricity in Finland", The Uranium Institute Twenty-fifth Annual Symposium, London (2000).

[2] FINNISH ENERGY INDUSTRIES FEDERATION FINERGY, Electricity Market 2015, Helsinki (2000).

Appendix 1

ENVIRONMENTAL IMPACT ASSESSMENT OF A NEW NUCLEAR POWER PLANT

Both Fortum and TVO launched environmental impact assessment (EIA) procedures in order to assess the environmental impacts of a new nuclear power unit, Loviisa 3 or Olkiluoto 3, respectively. The results of the environmental impact assessment on building or not building a new nuclear power unit were presented in an assessment report, where also the project alternatives were compared on the basis of their environmental impact. The EIA reports of Loviisa 3 and Olkiluoto 3 were submitted to the coordination authority, Ministry of Trade and Industry, in August 1999. The hearings required by the relevant legislation took place in autumn 1999, and the statement of the Ministry was given in February 2000. According to this statement, the EIA reports fulfil the valid requirements, which implies that the procedure is completed.

In the environmental impact assessment procedure, the environmental effects of the project and its alternatives were investigated and assessed. In addition to the EIA procedure, the building of a nuclear power plant in Finland requires the Council of State's decision in principle that has to be approved by the parliament and decisions on granting a permit in accordance with several acts, for example, the construction and operating licences conforming to the Nuclear Energy Act. The environmental impact assessment report that was compiled in the second stage of the EIA procedure is part of the decision-making material required in this procedure.

The purpose of the EIA procedure is to increase opportunities of the population to acquire information on the project and to affect it at such a stage when no binding decisions on the project have been taken. In the EIA procedure, background information, statements and opinions of the population are gathered for a decision in principle by the Council of State. The EIA procedure does not replace other studies or licences necessary for implementation of the project.

Three sets of project alternatives were assessed both at Loviisa and Olkiluoto:

- ”Main alternative” was the building of a new nuclear power unit: Either a PWR or BWR plant as well as several alternatives for the location and the cooling water intake and outlet were considered;
- ”Zero alternative” was the non-implementation of the project: The electricity would have to be purchased from other producers in Finland or abroad;
- Other alternatives allowed the generation of the corresponding amount of electricity by using coal, natural gas, peat, wood, hydroelectric power, wind power or solar panels.

The environmental effects of the project alternatives were investigated by comparing the changes they would cause to the present state of the environment, and these included the impact on

- Human health, living conditions and amenity;
- Soil, water, air, climate, organisms and interaction between them, and biological diversity;

- The community structure, buildings, landscape, townscape and cultural heritage;
- The utilisation of natural resources.

Most of the effects of a nuclear power plant are local and confined to the plant area and its vicinity. The effects of emissions of radioactive substances were assessed within a radius of about 10 km of the plant, whereas in the assessment of accidents, the inspection area was extended to 100-300 km. The impact of the cooling water was assessed within assessed within a radius of about 10 km. The social and socio-economic effects were assessed in the neighbouring municipalities. The environmental impact was investigated throughout the entire life cycle, from the building of the plant to its decommissioning. In addition, the possible procurement sources of nuclear fuel and the effects of the final disposal as well as the power transmission connection to the main grid were subjects of investigation.

Appendix 2

HIERARCHY OF LICENSING REQUIREMENTS

To assure the safety, licensability and well balanced design of the NPP concept, four different categories of regulations, marked with A to D according to descending priority in the following text, have to be taken into account in the design process. Especially in category C, there is a further internal hierarchy between the different national sets of nuclear safety guides.

A. Finnish legislation on nuclear energy

Supervision of the use of nuclear energy in Finland is mainly based on the

- Nuclear Energy Act (March 1, 1988)
- Nuclear Energy Ordinance (March 1, 1988)
- Radiation Act (January 1, 1992)
- Radiation Ordinance (January 1, 1992)
- Resolution of the Ministry of Social Affairs and Health (Nov. 5, 1968/594)
- Resolutions of the Council of State on the Safety of Nuclear Installations (No. 395/91, 396/91 and 397/91).

Compliance with Finnish legislation is mandatory.

B. Basic regulations

The ranking of the basic regulations is as follows:

- IAEA Safety Series No. 75-INSAG-3: Basic Safety Principles for Nuclear Power Plants
- US NRC General Design Criteria (10CFR50 Appendix A) as interpreted in the US NRC Standard Review Plan.

These sets of basic safety criteria reflect the common accepted ideas of nuclear safety, without going too long into technical details. Deviations from these criteria are acceptable only in exceptional cases. These deviations must be thoroughly documented and discussed with the national authorities at an early stage of the design process.

C. Sets of nuclear safety guides

Technical guidelines shall be applied in the following order of priority:

- YVL guides series 1-8
- European Utility Requirements (EUR)
- Appendices B, G, H, J and K to 10CFR50
- 10CFR50.46
- Applicable US NRC Regulatory Guides or KTA Guides.

Deviations from YVL guides are possible if the same standard of safety is attained. However, deviations must be defended with sufficient evidence. They should be kept to a minimum.

In case a certain matter has not been addressed in the YVL guides, other sets of guides shall be searched for this matter in the order given above. Hereby, caution is needed to make sure that the context is taken into account if a technical requirement is picked from a guide with lower priority. A mixture of loose, detailed requirements from different guides does not produce a sound and balanced level of safety.

D. Technical guidelines

The YVL guides in series 3 to 8 deal with different technical areas including mechanical engineering, construction, electrical engineering and I&C. However, the YVL guides are not detailed enough to provide the only guidance for the technical design of plant systems. Therefore, they have to be complemented with more specific engineering standards.

The standards to be applied to the design of a nuclear power plant have to be commonly accepted and widely acknowledged. Examples of such standards are KTA, DIN, ASME (mechanical engineering) and IEEE/IEC standards (electrical engineering and I&C). For each item, a consistent set of technical guidelines must be used.

Appendix 3

LIST OF CONTENTS OF THE DOCUMENT "SELECTED PRELIMINARY GENERAL TECHNICAL REQUIREMENTS FOR STUDYING THE FEASIBILITY OF A POSSIBLE NEW NUCLEAR POWER PLANT UNIT TO FINLAND"

1 Introduction

2 General design criteria

- 2.1 Plant type
- 2.2 Plant output
- 2.3 Plant site
- 2.4 Plant cooling
- 2.5 Technical lifetime
- 2.6 Radioactive release limits during normal operation
- 2.7 Occupational doses

3 External events and site conditions

- 3.1 Earthquake
- 3.2 Aeroplane crash
- 3.3 Missiles
- 3.4 External explosions
- 3.5 Lightning
- 3.6 EMI (Electromagnetic Interaction)
- 3.7 Airborne effluents
- 3.8 Blockage of the cooling water intakes
- 3.9 Wind loads
- 3.10 Snow and ice
- 3.11 High and low sea water levels
- 3.12 Sea water temperature
- 3.13 Air temperature
- 3.14 Ground water protection

4 Grid requirements

- 4.1 Connections to the external grid
- 4.2 Normal grid conditions
- 4.3 Short-term disturbances
- 4.4 Design considerations

5 Power plant characteristics

- 5.1 Normal power control
- 5.2 Power control during disturbances
- 5.3 Daily and weekly load cycling
- 5.4 Special operating cases
- 5.5 Fuel cycle flexibility

Appendices

Appendix 4

EUROPEAN UTILITY REQUIREMENTS (EUR)

The major European electricity producers have formed an organisation to develop the European Utility Requirement (EUR) document. The main objective of the EUR organisation is to produce a common set of utility requirements, endorsed by the European electricity producers, for the next generation of LWR nuclear power plants. Started with five partners in 1992, the EUR organisation now includes companies from ten countries: Belgium, Finland, France, Germany, Italy, Netherlands, Spain, Sweden, Switzerland, and United Kingdom. Russia has been accepted as an associated member since late 1998 with a view to become full member soon.

The utility requirements are addressed to the designers and suppliers of LWR plants. The aim of the requirements is to promote the harmonisation of

- 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 assessment of safety, reliability and cost and some of the corresponding criteria,

thus allowing the development of standard designs that can be built and licensed in several European countries with only minor variations.

Benefits are expected in two fields:

(1) improvement in the licensing of new nuclear power plants and in their public acceptance
- by setting common safety targets which are consistent with the best European and international objectives,
- by promoting within Europe common technical responses to safety problems,
- by setting "good neighbour" requirements, like low targets for accidents and routine radioactive releases into the environment, and consideration of decommissioning aspects at the design stage;

(2) strengthening of nuclear electricity competitiveness
- by controlling construction costs and operating costs through standardisation, simplification and optimisation of maintenance at the design stage,
- by establishing stable conditions for competition between the suppliers on the European Market
- by allowing low operation and fuel cycle costs, through flexible and efficient design features that allow the easy adaptation to future plant operating and fuel management schemes,
- by laying down ambitious (but achievable) availability and lifetime targets.

The EUR document is structured into four volumes. Each volume is divided into chapters that deal with a specific topic. The chapters are subdivided into sections.

Volume 1 Main policies and objectives
This volume presents the major objectives of the EUR organisation and the main policies, which are implemented in the EUR document. It also summarises the most important requirements developed in Volumes 2 and 4.

Volume 2 Generic nuclear island requirements
This volume contains all the generic requirements and preferences of the EUR utilities for the nuclear island which are not related to any specific design.

Volume 3 Application of EUR to specific designs
This volume consists of a number of subsets. Each subset is dedicated to a specific design that is of interest to the participating utilities. It contains a description of the standard nuclear island, a summary of the analysis of compliance vs. Volumes 1 and 2 and, where needed, design dependent requirements and preferences of the EUR utilities. It also includes the information related to that design called for in certain requirements of Volume 2.

Volume 4 Power generation plant requirements
This volume contains the generic requirements related to the power generation plant.

The EUR document is published in successive stages called revisions. The first stage of the publication, Revision A, was released in March 1994. The second stage, Revision B, took place in late 1995. The latest version of Volumes 1 and 2, Revision C, is scheduled for publication in 2001. By the end of 2000, 3 subsets of Volume 3 have been released:

- subset 3A dedicated to BWR 90 (developed by ABB Atom)
- subset 3B dedicated to EPR (developed by NPI, Framatome and Siemens)
- subset 3C dedicated to EP 1000 (developed by Westinghouse and Ansaldo).

Two other subsets are being produced that are dedicated to ABWR (by General Electric) and SWR 1000 (by Siemens). Additional subsets of Volume 3 will be produced by mutual agreement.

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Appendix 5

IAEA-TECDOC-1123
"STRATEGIES FOR COMPETITIVE NUCLEAR POWER PLANTS"

The following strategies with related techniques are given for new plants or projects that are being considered:

1 *Make efficient design choices*

1.1 Make the most profitable choice among fuels and technologies that is possible
1.2 Recognise that plant design can affect profitability under different market conditions

2 *Secure all economic risks and liabilities efficiently*

2.1 Allocate the risks of plant construction and operation efficiently, and reward the party that is accepting risk
2.2 Specify the allocation of plant completion risks, including allocation of and terms for risks due to policy change
2.3 Establish liability for political and policy based risks during operations
2.4 Allocate safety risks among operations and plant design areas, establish patterns of liability for safety risks
2.5 Potential domestic and foreign liabilities from plant operations and failures must be specified and allocated among plant managers, governments, and appropriate agencies
2.6 Responsibilities and funding for waste fuel management, retired plant disposal, and other perceived open-ended liabilities after plant closure must be clearly allocated

3 *Focus on profitability*

3.1 Meet financial criteria for net returns and for risks in order to assure the availability of funding
3.2 Design construction plans to minimize the net financial effects of interest during construction and delay
3.3 Design the plant to reduce the net costs of down time from operational failures
3.4 Reduce capital costs

4 *Know and serve your markets*

4.1 Design the plant to meet the capabilities and interests of the electricity grid and potential market structures
4.2 Identify and develop market niches for the power plant
4.3 Consider revenue sources other than power generation (cogeneration, desalination, etc.)

Attachment A

R. TARJANNE, S. RISSANEN, LAPPEENRANTA UNIVERSITY OF TECHNOLOGY "NUCLEAR POWER LEAST-COST OPTION FOR BASE-LOAD ELECTRICITY IN FINLAND"

The following conclusions are included in the report presented in the Uranium Institute Annual Symposium in September 2000.

The possible options for new base-load power generation in Finland studied were as follows:

- nuclear power plant,
- combined cycle gas turbine plant,
- coal-fired condensing power plant and
- peat-fired condensing power plant.

The existing 560 MW Meri-Pori power plant with pulverised coal combustion has been used as the reference unit for the coal-fired power plant. The peat-fired unit is based on fluidised bed combustion. The performance and cost data of the combined cycle gas turbine plant is based on new efficient concepts now available internationally.

The sizing of the gas- and coal-fired units has been selected so large that the scale benefit can be utilised as far as possible. The coal plant would be located on the seacoast. The size of the peat plant is restricted to 150 MW, because the transport distance of peat fuel is growing too long for bigger unit sizes.

The sizing of the nuclear alternative is selected in the middle of the range of the reactors under consideration. The investment and operation costs of the nuclear unit are based on the fact that it would be built on an existing nuclear site. The construction time of the nuclear power plant is supposed to be five years. All the expenses of nuclear waste treatment (including spent fuel) and decommissioning of the plant are included in the variable operation and maintenance costs through the annual payments to the nuclear waste fund.

The annuity method has been applied for calculating the electricity generation costs of the four alternatives. A real interest rate of 5 per cent per annum and the fixed price level of February 2000 have been used. Based on these assumptions, the nuclear power plant has the lowest electricity generation cost, when the utilization time exceeds 6100 hours corresponding to a capacity factor of 70 per cent.

The electricity generation costs of the four alternatives with the annual full-load utilization time of 8000 hours (corresponding to a capacity factor of 91 per cent) are as follows:

- nuclear electricity: 22.3 EUR/MWh
- coal based electricity 24.4 EUR/MWh
- gas based electricity 26.3 EUR/MWh
- peat based electricity: 31.3EUR/MWh.

The capital cost component is dominating in the nuclear generation cost, whereas the nuclear fuel cost remains quite low. For the other alternatives under consideration, the fuel cost component is highly dominating.

The sensitivity analysis reveals that the advantage of the nuclear option is quite insensitive for the changes of the input parameters. E.g. the growth of the uranium price causes only a slight increase in the nuclear electricity cost, whereas for the natural gas alternative the rising trend of gas price causes a major risk. Furthermore, the availability of natural gas in Finland for a new big base load unit is not guaranteed in the near future.

Based on the financial comparison the nuclear alternative is the least-cost option for new base-load capacity in Finland. The nuclear electricity would cost 22.3 EUR/MWh, having the margins of 2 EUR/MWh and 4 EUR/MWh compared to coal- and gas-based electricity, respectively. The sensitivity analysis reveals that the nuclear option is rather insensitive to the changes of the input data, whereas the gas alternative involves a considerable risk as a consequence of increasing gas price.

Of the four alternatives under consideration, the nuclear option is the only one, which does not generate any carbon dioxide emissions to the atmosphere. A new 1250 MW nuclear unit with 10 TWh annual production would save 8.3 million tons carbon dioxide emissions annually, if the reference is the coal-fired condensing power plant. Compared to the combined cycle gas turbine plant, the new nuclear unit would save 3.7 million tons carbon dioxide emissions, respectively. The nuclear choice would make a major contribution for achieving in 2010 the greenhouse gas emission level in accordance with the Kyoto protocol.

From the national point of view – both in terms of economy and in terms of the Finnish compliance with its Kyoto protocol commitments on greenhouse gas emissions reductions – the nuclear choice is by far the best alternative for new base-load power capacity.

Attachment B

FINERGY REPORT "ELECTRICITY MARKET 2015"

The Finnish Energy Industries Federation Finergy prepared a report in 2000 where the developments in the use and procurement of electricity in Finland and Nordic countries are examined up to year 2015. The following is an extract of the conclusions.

The programme of the Finnish Government outlines the country's economic policy so that Finland has to provide an internationally competitive operating environment for capital and corporate operations. Finland's energy policy focuses on energy economy as a whole, on the various energy sources and forms and on their mutual proportions. The specified areas of action include the promotion of the energy generation structure towards an energy balance containing a smaller proportion of coal, promotion of the energy market, efficiency and energy conservation, promotion of domestic energy, ensuring a sufficiently versatile and inexpensive energy supply capacity, and maintaining supply reliability.

This report examines the opportunities of the Finnish energy industry to respond to the challenges presented in the programme of the Finnish Government and also to challenges having a longer perspective. Planning the future of the energy industry is long-term work, and the decisions made will have an impact extending over decades. We must have our eyes on the new century and millennium.

The electricity market is developing through the Nordic countries and its adjacent areas towards other parts of Europe. The Nordic electricity market is already integrated to a relatively high degree. In the EU countries, the opening up of the electricity market will be a fact during the time span of this report. In Finland, it is the responsibility of us working in the energy industry to make sure that as far as energy supply is concerned, our industries and society can develop further.

The consumption of electricity in Finland is estimated to grow from the current almost 80 TWh per year to almost 100 TWh by the year 2015. The consumption prognosis, indicating an annual growth of 1.3 per cent, has been drawn up on the assumption that economically viable electricity conservation options have been implemented and that wherever economically possible, the use of electricity has been intensified.

The Finnish power plant capacity is in efficient use. The importance of thermal power capacity is highlighted during years when there is but little water available for hydropower generation. The electricity generation capacity in the other Nordic countries rests to a great extent on hydropower and consequently depends on rainfall. Finland will need a lot of new power generation capacity by 2015. Here, combined heat and power production will also continue to hold a key role.

Imports of electricity will not solve Finland's long-term electricity needs. Finland needs to prepare for a situation where even Sweden and Norway have to import an increasing proportion of their electricity. It is also likely that as the European electricity market opens up, the flows of electricity will start to run from the north to the south rather than from the south to the north. Price and environmental issues will be the decisive factors. The proportion of imported electricity of all electricity consumption in Finland is already so high that the energy industry cannot recommend increasing the imports further.

Increasing the proportion of fossil fuels in Finnish power generation is not a generally desired option. On the other hand, the use of new, renewable sources of energy will only solve a part of Finland's need for additional electricity generation capacity. However, we need to make decisions now while at the same time looking far into the future to a time when Finland's current power plants become outdated. We also need to make sure that we have reserve capacity in electricity generation in the short term.

In Finland and other Nordic and EU countries, it is difficult to fulfil the obligations stated in the Kyoto Protocol on climatic issues. In view of the present situation and policies, it can be reasonably asked whether the EU can meet its objective of reducing greenhouse gas emissions by 8 per cent. Alongside renewable energy sources, we must make extensive input in research and development and pursue the planning of time-consuming and low-emission technologies and systems. Despite this, it is difficult to see how this objective could be fulfilled without current and new nuclear power.

Relative electricity need in Finland has grown clearly faster than in any other Nordic country in the 1990s, being at the same level with the growth in Norway even on an absolute scale. In Sweden, the need for additional electricity has almost halted to a level of approximately 0.5 per cent annually. In Norway and Denmark, however, this growth rate has been approximately 1.5 per cent a year. The differences in the growth in electricity consumption in particular have been influenced by the rapid industrial development in Finland, improved standard of living and the widespread use of electric heating, which, however, is still more common in Norway and Sweden than in Finland.

Consumption of electricity in the Nordic countries is expected to grow from the approximately 377 TWh in 1999 to approximately 420 TWh in 2015. The biggest growth is still anticipated in Finland. Growth in industrial production will make the biggest contribution to the increase in electricity use in Finland. Industries will account for a higher proportion of electricity use in Finland than in the other Nordic countries. Similarly, households and the service sector will display slightly faster growth in electricity consumption than in the other Nordic countries due to urbanisation, smaller unit size of households and continued increase in the standard of living.

At the moment, more than half of all electricity generated in the Nordic countries is produced through hydropower. In a dry year, hydropower provides 74 TWh less electricity than in a year with a good water situation; this is almost as much as the entire consumption in Finland. Even during years with an average water situation, both Norway and Sweden import a lot of electricity.

Finland's electricity procurement options cannot be built on there being a continuous supply of inexpensive electricity available on the Nordic market. As the consumption of electricity increases, it can be expected that the Nordic price level, too, will rise. New investments in power generation require that the proceeds derived from the open market, varying in line with the water situation, are sufficient to cover the investments made.

The high proportion of hydropower and good regulation options in the Nordic market area mean that most of other energy generation – base load – has a long annual usage time. It is most lucrative to produce this power in power plants where the variable costs are small and whose production can be sold on the market at a profitable price. In Finland, these options are combined heat and power production and nuclear power.

Nuclear power: A competitive option?

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Abstract. Because the future development of nuclear power will depend largely on its economic performance compared to alternatives, the OECD Nuclear Energy Agency (NEA) investigates continuously the economic aspects of nuclear power. This paper provides key findings from a series of OECD studies on projected costs of generating electricity and other related NEA activities. It addresses the cost economics necessary for nuclear units to be competitive, and discusses the challenges and opportunities currently faced by nuclear power.

1. INTRODUCTION

Economic competitiveness is the cornerstone for the successful deployment of any electricity generation source and technology. Decisions on technologies and energy mixes for electricity generation have to take into account a variety of non-economic issues, including social, environmental and health impacts but utilities base their choices primarily on the costs of generating electricity from alternative energy sources and technologies available on the market.

The evolution of the policy making landscape, including economic deregulation and privatisation of the power sector but also an increasing awareness of sustainable development goals, leads to changes in the framework of economic assessment. This evolution creates new challenges and opportunities for different generation technologies, including nuclear power.

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

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

Since the future development of nuclear power will depend largely on its economic performance as compared with alternatives, the OECD Nuclear Energy Agency (NEA) investigates continuously economic aspects of nuclear power. The series of OECD studies on projected costs of generating electricity provides documented data and detailed analyses on the status and trends in generation costs. This paper is based upon key findings from those studies and other related NEA activities. It addresses the cost economics necessary for nuclear units to be competitive. The challenges and opportunities resulting from the new economic landscape, and the ways in which they might affect the competitiveness of nuclear power are discussed.

2. DIRECT LEVELISED COST COMPARISONS

The last OECD study on projected cost of generating electricity [1], published in 1998, covers cost data for baseload power plants that could be commissioned by 2005-2010 in the nineteen countries which participated in the study. Although the energy sources and technologies considered vary from country to country, the main alternatives are coal-fired, gas-fired and nuclear power plants. The scope of the study excludes hydro power plants because their costs are highly site specific. As far as other renewable sources are concerned, cost information on their use for electricity generation was provided by three countries only.

The OECD method and results are not a substitute for economic studies that would be carried out by utilities based upon detailed cost elements corresponding to a given project and taking into account the overall context of electricity system expansion. However, the outcomes are indicative of the relative competitiveness of alternative options and point to the most economically attractive options in each country.

Twelve countries provided cost information for at least one nuclear unit and one alternative. The cost estimates presented below are based upon that information. The levelised generation costs were calculated using cost elements provided by participating countries, a commonly agreed methodological framework, and generic assumptions for some key parameters. Generic assumptions include a 40 year lifetime and a 75% load factor for all power plants considered. For gas-fired power plants, the costs of replacing major equipment at the end of their technical lifetime, around 20 years, are included in the investment costs. The discount rate adopted to estimate levelised generation costs is a key parameter. The last OECD study used two real discount rates as reference, 5% and 10% per annum, that are considered representative of the range of values used by electricity producers in most countries. Fuel cost assumptions were provided by participating countries and, therefore, fuel costs in the commissioning year and fuel price escalation rates are country specific.

Cost elements were provided by each country in its national currency of 1 July 1996 but for consistency sake those costs were converted in dollar of the United States of the same date using official exchange rates prevailing at that date. Levelised costs, calculated in a unique currency, can be presented and compared in a consistent manner. Most countries provided data for several coal-fired, gas-fired or nuclear power plants, only the cheapest plant for each alternative is showed on Table I which summarises the main results of the last study.

The ranges of generation costs for each technology/energy source are quite broad (see Table I) showing that competitiveness should be assessed on a case by case basis at the country or utility level taking into account the specific technical and economic conditions applicable in each case. Nevertheless, average generation costs for each technology/source are indicative and the cost ratios in each country illustrate the ranking of alternative options.

TABLE I. PROJECTED LEVELISED GENERATION COSTS (USMILL OF 1.7.1996/KWH)

	At 5% discount rate			At 10% discount rate		
Country	Coal	Gas	Nuclear	Coal	Gas	Nuclear
Canada	29.2	30.0	24.7	37.0	33.0	39.6
Finland	31.8	35.9	37.3	39.1	41.1	55.9
France	46.4	47.4	32.2	59.5	53.3	49.2
Japan	55.8	79.1	57.5	76.1	84.4	79.6
Korea	34.4	42.5	30.7	45.0	47.0	48.3
Spain	42.2	47.9	41.0	54.7	54.4	63.8
Turkey	39.8	30.7	32.8	48.7	33.9	51.8
United States	25.0	23.3	33.3	34.7	23.6	46.2
Brazil	35.4	28.5	33.1	43.2	32.7	46.7
China	31.8	n.a.	25.4	40.0	n.a.	39.0
India	33.0	n.a.	32.8	40.2	n.a.	51.0
Russia	46.3	35.4	26.9	55.3	39.0	46.5

On average, projected generation costs for coal-fired power plants are around 38 mill/kWh[1] at 5% discount rate and around 48 mill/kWh at 10%. Those costs are based upon coal prices ranging from 1 \$/GJ to 2.8 \$/GJ in 2005 – year of commissioning of the plant – and increasing at an average escalation rate of 0.3% per annum. For gas-fired power plants, the average projected generating costs are 40 mill/kWh and 44 mill/kWh at 5% and 10% discount rate, respectively. The gas prices assumed vary between 1.6 \$/GJ and 5.4 \$/GJ in 2005 with a 0.8% per annum average escalation rate. The average generation costs for nuclear power plants are 34 mill/kWh and 51 mill/kWh respectively at 5% and 10% cent discount rate. This shows that nuclear power has the potential to compete favourably at 5% discount rate but looses most of its competitive margin at 10% discount rate.

[1] 1 mill = 10^{-3} \$.

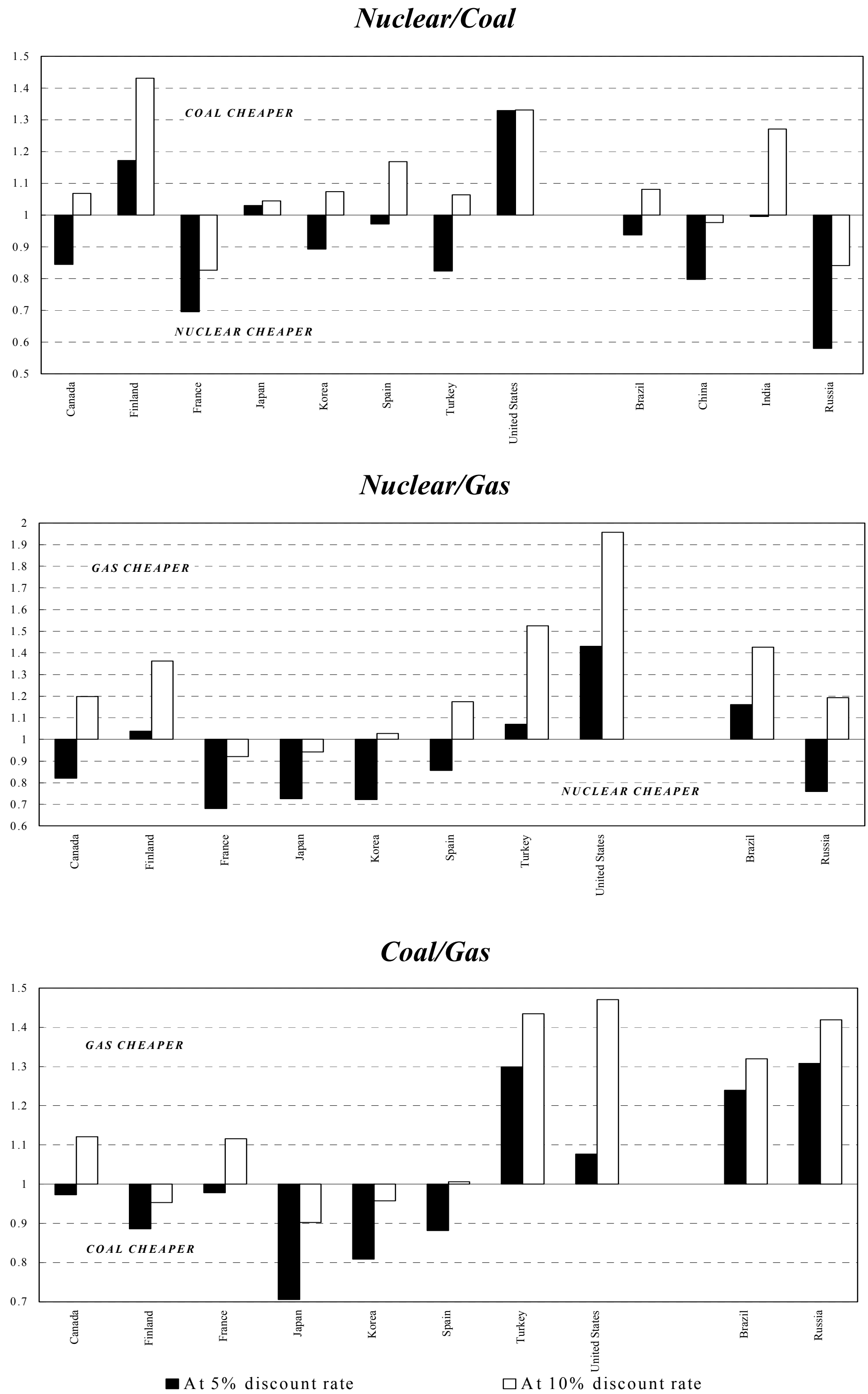

FIG. 1. Generation cost ratios.

In the twelve countries where coal and nuclear options are considered, the ratios between projected costs of nuclear and coal generated electricity range from 0.58 to 1.33 at 5% discount rate and from 0.73 to 1.43 at 10% discount rate (see Figure 1). In the ten countries where gas and nuclear options are considered, the ratios between projected costs of nuclear and gas generated electricity range from 0.68 to 1.43 at 5% discount rate and from 0.92 to 1.96 at 10% discount rate. In the same countries, the ratios between projected costs of coal and gas generated electricity range from 0.71 and 1.31 at 5% discount rate and from 0.9 and 1.47 at 10% discount rate. The analysis of key results from the study shows that nuclear can be the cheapest option in countries where capital costs of nuclear power plants can be kept low, and where gas and/or coal prices are rather high and are projected to increase during the economic lifetime of the plants.

Owing to uncertainties on projected cost elements and to the conceptual level of detail inherent to international studies based upon generic assumptions, small differences in generation costs may not be significant but differences higher than 10% may be considered as indicative of the relative competitiveness of alternative options in each country. Within the twelve countries that provided data for nuclear power and at least one other option, at 5% discount rate, nuclear is the cheapest by a margin of at least 10% in five countries, coal is the cheapest by a margin of at least 10% in one country and gas is the cheapest by a margin of at least 10% in one country. At 10% discount rate, nuclear is the cheapest option by a margin of at least 10% in no country, coal is cheapest option by a margin of at least 10% in no country and gas is the cheapest option by a margin of at least 10% in five countries. This confirms the difficulties for nuclear power to compete at high discount rates.

Costs of generating electricity have decreased continuously during the last decade or so owing to technology progress, more efficient plant management and lower fuel prices. The results from the series of OECD studies on projected costs of generating electricity illustrate those trends for coal-fired, gas-fired and nuclear power plants. Similarly, renewable energy sources, although they remain expensive for electricity generation and are seldom competitive, have experienced drastic cost reductions recently. Regarding trends in cost ratios, one of the main findings from the last two studies in the series is the rapidly increasing competitiveness of gas for base-load generation. Modern gas-fired combined cycle power plants having high efficiency, 50% or more, and low capital intensity, are challenging the competitiveness of coal and nuclear at present gas prices on international markets.

In spite of the relevance of economic comparisons, it is important to be aware of their limits. The input values for any evaluation of generation costs, whether they are tied specifically to a project or are generic in nature, are not known with absolute certainty. Fossil fuel prices, regulations, environmental standards, and other factors may change from what were originally expected. The uncertainty of input values generally leads decision makers to take into account potential variations in the values of some cost factors and they may also look beyond direct costs in their evaluations.

3. NUCLEAR GENERATION COST STRUCTURE

For a nuclear unit, over half of the total generation cost is related to capital investment while for coal and gas, fuel represents some 40 to 80% of the total generation cost. The high capital costs of nuclear power plants hamper their competitiveness, especially at high discount rates and nuclear plant investors must accept long periods of time for return of their invested capital in order to attain competitive generation costs. Therefore, reducing investment costs is a prerequisite for enhancing the competitiveness of nuclear power. A recent NEA study [2] analyses means to reduce the capital cost of nuclear power plants, identifying as the most significant: plant size, multiple unit sites, design improvement, standardisation, modularisation and performance improvement.

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

TABLE II. OVERNIGHT COSTS OF NUCLEAR POWER PLANTS, NORMALISED TO 1.0 FOR 1 × 1000 MW(E) UNIT

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

* Reactor size in MW(e)

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

The capital costs of the Korea Standard Nuclear Power Plant (KSNP), a 1000 MW(e) PWR, show a similar trend. Today, one KSNP unit is in operation and five more units are under construction. Table III illustrates expected capital cost reductions of subsequent KSNPs, based on contract prices.

TABLE III. CAPITAL COSTS OF SUBSEQUENT KSNPS, NORMALISED TO 1.0 FOR 1ST & 2ND UNITS

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

Following the KNSP, Korea has started a programme for the development of the Korea Next Generation Reactor (KNGR), a 1 300 MW(e) PWR. The key objective of the KNGR development programme is to enhance safety and economics. The cost reductions as compared with existing KSNP are shown in Table IV which indicates the main factors leading to those reductions. Globally, the new generation of plants is designed to be around 17% cheaper than previous nuclear units.

The bottom line is, however, that past experience and recent evaluations point to a maximum potential reduction of capital costs for nuclear units by 25% will probably not be enough to secure economic competitiveness with fossil-fuelled power plants.

TABLE IV. EXPECTED CAPITAL COST REDUCTION COMPARED WITH KSNP AND INFLUENCING FACTORS

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

Operations and maintenance (O&M) costs represent a relatively small component of the total generation cost for nuclear power plants, although in some countries they exceed fuel costs. At 10% discount rate, they represent some 15% of the total cost of nuclear generated electricity in most countries and at 5% discount rate, the share of O&M cost generally reaches or exceeds 20%. The O&M costs are influenced by technical performance of the nuclear power plants and, moreover, by safety regulations and manpower costs prevailing in different countries. Therefore, they vary significantly both in absolute and relative value from country to country. The reasons for the wide disparity in O&M costs in different countries have been analysed in an NEA study [3] which concluded that international cost comparisons are difficult owing to the major role of country specific factors in these costs and to the lack of harmonised methodology for calculating O&M costs.

In the past, escalation in O&M costs has been mainly due to regulatory factors and, to a lesser extent, to the increasing cost of manpower. Lowering or at least stabilisation of O&M costs has been experienced recently through learning from increasing experience in operating a growing number of nuclear power plants and reaching stable regulatory procedures. Also, in countries where the electricity sector has been deregulated already, more efficient management methods have been introduced that lead to lower O&M costs. Moreover, advanced reactor designs have simplified operations and maintenance process as well as enhanced performance, leading to an overall reduction of O&M costs.

Nuclear fuel accounts for less than one quarter or less of total generation cost. In contrast, fuel can account for one half of coal-fired generation cost and three quarters of gas-fired generation cost. In light of the small proportion of the total generating cost taken up by the nuclear fuel cycle component, nuclear generation costs are relatively insensitive to uranium and fuel cycle service price volatility. However, decreases in fuel cycle costs experienced during the last years have contributed significantly to the overall trend in nuclear electricity generation cost decrease.

In the recent years, fuel cycle costs have decreased significantly for all types of nuclear power plants in all countries. Technical improvements leading to efficiency gains have led to a reduction in the costs and prices of most nuclear fuel services. According to the NEA studies on economics of the fuel cycle [4], a 40% real term reduction in estimated lifetime levelised nuclear fuel cycle costs has occurred since 1985. This reduction is due to improved reactor and fuel performance and lower prices of uranium and some fuel cycle services. Improved fuel and reactor performance factors contributed some 20% of the total reduction in nuclear fuel cycle costs. Major decreases in the prices of uranium and enrichment services, and reduction in back-end service prices contributed 80% of this reduction.

The most important technical factors that have an impact on nuclear fuel cycle costs are the burn-up in reactors and tails assay of enrichment plants. The discount rate has little influence on the total fuel cycle costs. The levelised costs of front-end steps are increasing with the discount rate while for the back-end steps, in particular spent fuel or high level waste disposal, increasing the discount rate decreases levelised costs since these operations occur after electricity generation.

The downward trend in uranium prices that occurred since the late seventies has contributed significantly to the reduction of fuel cycle costs. Drastic uranium price escalation does not appear very likely in the short term owing to the existing excess inventories of fissile materials. In the long term, even if uranium prices were to rise either by market mechanisms or by increase in the production costs, the effect on the total nuclear fuel cycle and electricity generation costs would be limited. A doubling of the uranium price would lead to only some 20% increase in the nuclear fuel cycle cost.

Enrichment prices decreased by some 30% between 1985 and 1990. This trend is expected to continue owing to efficiency improvement in the existing enrichment facilities and to market forces as long as supply capabilities will excess demand. In the longer term, the enhancement of presently used technologies and the possible entry on the market of new processes should lead to cost and price reduction for enrichment services.

Reprocessing costs are expected to decrease through learning from experience and efficiency gains as new industrial facilities are commissioned and the overall process reaches commercial maturity. The same is applicable to direct disposal of spent fuel.

4. CONCLUDING REMARKS

Today, a new nuclear power plant is seldom the cheapest option. With the power sector undergoing deregulation and privatisation, the market is not the most attractive for highly capital intensive technologies such as nuclear power. In the present and expected future business environment, risks will be greater for investors embarking on nuclear projects while technologies with lower capital cost will have an advantage.

In a free deregulated market, economic competitiveness is a key factor, if not the only one, in selecting an option. Existing nuclear units, when they are well operated and managed, generally have a clear economic advantage owing to their low marginal cost. New reactors, on the other hand, will have difficulties to compete. They must achieve significantly lower capital costs per installed capacity, and total generation costs lower than alternatives to be successfully deployed. This may be difficult to accomplish as nuclear technologies seem inherently capital intensive and fossil fuel prices are currently low and are projected to rise only modestly.

However, nuclear power may regain the competitive margin that it enjoyed in the mid-70s. There are innovative reactor concepts, such as the modular high temperature gas cooled reactor or the new generation of Korean PWRs, that might eventually meet the objective of very low specific capital costs. Moreover, in some countries indigenous fossil fuel supplies are scarce and/or expensive, and in these areas, nuclear power is likely to keep an economic advantage.

The experience already acquired in some countries has shown the potential to further decrease the costs of nuclear power generation through decreasing investment, operation and maintenance, and fuel cycle costs. Nuclear power plant costs can be minimised if certain conditions exist at the outset. These conditions include design simplification and standardisation, clear and stable regulatory requirements, a high fraction of design completion before construction, use of multiple unit sites with phased construction, and use of modular construction.

A factor that may influence the competitiveness of nuclear power in the future is external costs. National policy issues related to energy security and diversity of supply may modify the selection process from one of pure relative internal cost. In addition, the global environmental impacts of various power generation technologies are not completely internalised at present. Should this occur, nuclear power will likely have an improved economic ranking world-wide. It is not clear, however, if and when such recognition will take place.

The challenges for nuclear energy are: to secure and demonstrate the competitiveness of units currently in operation through efficient operation and management and continued reduction of fuel cycle costs; and to develop a new generation of reactors that could successfully secure competitiveness when existing units will have to be replaced.

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Development of new nuclear power plants in the Republic of Korea

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Abstract. Nuclear power in Korea is one of our major energy sources, which accounting for approximately 50 % of the total share of electric generation using the safest and most stable methods. Based on the outstanding performance of nuclear power generation, Korea plans to construct eight (8) new nuclear power plants to maintain nuclear power as a major contributor to the national energy mix by 2014. In order to ensure that nuclear power plants are safer and more economical than any conventional electric power sources, KEPCO has developed the improved Korea Standard Nuclear Power Plant ($KSNP^+$) and the Korea Next Generation Reactor (KNGR) by utilizing over 30 years of expertise and learned technologies gained from construction, design, operation of sixteen nuclear units. Recently, KEPCO has developed its own project management tool, the Nuclear Project Control System (NPCS), which integrates schedule, material, cost, drawing and documentation into a computerized system, to be utilized for construction of the nuclear power plants. This paper summarizes KEPCO's various efforts for design improvement of $KSNP^+$ and KNGR in terms of performance and economic viability for construction of new nuclear power plants in Korea.

1. INTRODUCTION OF KOREAN STANDARD NUCLEAR POWER PLANT

Since 1984, KSNP has been developed through nuclear power plant standardization projects by incorporating the latest technologies and the expertise gathered during years of construction and operation of nuclear power plants in Korea. Ulchin unit 3 is the first of the KSNP design and has been in commercial operation since 1998, demonstrating an outstanding operating record in performance and safety. The adoption of the KSNP by KEDO, as a reference plant for the Sinpo Nuclear Power Plant under construction in North Korea, demonstrates the design acceptance by the international nuclear industry.

The characteristics of KSNP can be explained with respect to the significant design improvement accomplished by: (1) applying state-of-the-art technology to the extent that is justifiable, based on proven technology, (2) implementing design simplification and optimization and (3) considering the human factor engineering, which results in improved plant safety and performance with additional operating margins and improvements in constructability and maintainability.

KSNP is the newest nuclear power plant based on proven advanced technologies. The features are as follows:

- Application of Safety Depressurization Systems;
- Increased safety during Shutdown and Mid-loop Operation;
- Reduced probability for Loss of Coolant Accident;
- Application of Leak Before Break (LBB) design concept in the Reactor Coolant System piping, Shutdown Cooling System, Safety Injection System, and Pressurizer Surge Line;
- Incorporation of Human Factors Engineering concept to design the Main Control Board;
- Improved Operability, Maintainability, and Accessibility;
- Reliability improvements of Plant Electrical System;
- Separation between redundant trains of Safety Related System;
- Application of Passive Flood Protection Design.

The safety of the KSNP design is assured by following concepts of defense in-depth such as prevention of accidents or departure from normal operation, early detection by monitoring system, mitigation of small accidents to prevent the progression to severe accidents, etc. The design concept for plant safety consists of safety related design, safety related system and components and safe plant operation thus achieving the safety goal. To prevent excessive release of radioactive materials to the environment, the KSNP design provides multiple barriers including nuclear fuel, fuel cladding, reactor coolant pressure boundary and finally containment building.

2. STATUS OF THE IMPROVED KSNP (KSNP$^+$) DEVELOPMENT

2.1. Background of Development

The economic, technical efficiency and localization rate were enhanced by repeated application of the KSNP design to Yonggwang 5&6 and Ulchin 5&6 after Ulchin 3&4, the first 2 units of the newly designed KSNP series.

The design improvement and reduction in construction quantity to enhance the plant economy, however, were limited in depth due to the lack of flexibility in project schedule. This can be seen through the Korean construction practice that Ulchin 3&4, Yonggwang 5&6 and Ulchin 5&6 have been successively constructed with only one or two years time difference. Additional reductions in the construction costs are not expected, as cost savings resulting from the repeated construction of KSNP has reached a critical point.

The nuclear power program in Korea, now, is confronted with a difficult situation by losing a comparative advantage in the generating cost over other power sources.

The recent international economic environment has entered into a boundless competition era due to the initiation of the WTO regime and complete market opening of each country; the international nuclear industry has been inevitably exposed to strong competition.

To cope with these environmental changes, it is inevitable that the design concept of the existing KSNP should be re-established based on Koreas technical capability, accumulated construction and operation experiences and new innovative nuclear power plant model. That is, it is requested to develop more internationally marketable “improved KSNP”, rather than partial design improvements of the existing KSNP series and enhancing the safety and economy of the KSNP design by incorporating reformative and comprehensive improvements.

2.2. Major design improvement

The design improvement concepts to develop the KSNP$^+$ are as follows:

– Optimization and simplification of System/Facility/Structure by reflecting the operating experience of the existing nuclear power plants;
– Optimization of building volume and construction material by optimizing the plant building and equipment layout;

Application of improved design concepts through the review of the design concepts of the reference plant;

- Application of the advanced design of other type nuclear power plants;
- Optimized design considering local site characteristics;
- Incorporation of expertise and recommendations of constructors and vendors;
- Application of foreign state-of-the-art technology and construction methodology.

The following is a major design improvement of the KSNP$^+$ compared with the original KSNP design.

2.2.1. Optimization of plant arrangement

- Designed a single Compound Building to include non-safety-related buildings (Auxiliary Bldg., Access Control Bldg., and Radwaste Bldg.);
- Eliminated underground Radwaste Tunnel and minimize of the length of the Underground Common Tunnel;
- Minimized the piping, cable tray and HVAC duct lengths;
- Reduced building volume and construction material;
- Reduced occupational radiation exposures by enhancing operability and maintainability.

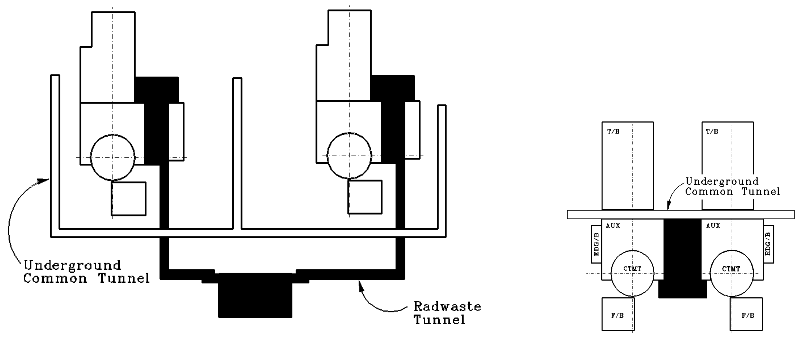

Figure 1. Optimization of general arrangement.

2.2.2. System design optimization

- Combined Plant Monitoring System (PMS) and Plant Annunciator System (PAS) into a Plant Monitoring & Annunciator System (PMAS) to eliminate redundant peripherals and human factor engineering inconsistencies;
- Chemical and Volume Control System (CVCS) Optimization;
- Optimized the capacities of larger CVCS Tanks and Letdown Heat Exchangers;
- Adjusted rational safety and quality classes based on ANSI 51.1;
- Eliminated RCP Seal Injection Heat Exchangers;
- Reduced the number of Circulating Water System (CWS) Pumps and Travelling Screens (6 to 4 per unit).

2.2.3. Optimization of equipment capacity and application of advanced technology

- Reduced capacity of Emergency Diesel Generator, Auxiliary Boiler and large capacity pumps;
- Replaced the Active Hydrogen Recombiner of the Containment Hydrogen Control System with a Passive Autocatalytic Recombiner (PAR);
- Improved Control Method for Plant Control System
  - Changed from Single-Loop to Multi-Loop control concepts
  - Added control functions to Local Multiplexer;
- Adopted Steel-Concrete Composite Structure
  - Improved construction area availability by reducing structural member sizes
  - Reduced construction material requirement
    - Eliminated embedded plates on the ceiling
    - Reduced the quantity of temporary construction structures;
- Applied Area Completion Concepts and Deck Plate Construction Method;
- Applied jetty, access pit, and modularization.

2.2.4. Improvement of operability and maintainability

- Designed Integrated Reactor Vessel Head Assembly (IHA);
- Replaced temporary-type Refueling Pool Cavity Seals with permanent-type ones;
- Changed the design of the Ex-core Neutron Flux Monitoring System with long life Fission Chamber-type detectors;
- Optimized the number and extension of the service life of In-core Instrumentation.

2.3. Advantages of design improvement

The effects of the design improvement are expected as follows:

- Increase of reliability of the systems important to monitor plant status and decrease of chances of human error by the simplified system design;
- Enhancement of operability, maintainability, and accessibility of plant workers by virtue of simplification of design and plant arrangement optimization;
- Reduction of building volume, optimization of plant site area, and enhancement of constructibility by optimization of design and adoption of new technology;
- Reduction of construction and maintenance cost.

3. DEVELOPMENT OF THE KOREAN NEXT GENERATION REACTOR (KNGR)

3.1. Background

In order to further enhance safety and economic competitiveness, a new project to develop an ALWR called KNGR, 1400 MWe PWR, was launched in 1992. Like other ALWRs being developed worldwide, KNGR reflects operating experiences as well as the technology accumulated through the KSNP design. Also, the development of KNGR is closely linked to the construction plan so that the design can be realized in due time.

The KNGR is an evolutionary ALWR based on the current Korean Standard Nuclear Power Plant (KSNP) design with capacity increment. It also incorporates a number of design modifications and improvements to meet the utility's needs for enhanced safety, economic goals and to address the new licensing issues such as mitigation of severe accidents.

3.2. Major design characteristics

- System thermal power of 4,000 MWth with electric power of 1,400 Mwe;
- Containment
  - Single pre-stressed concrete with steel liner
  - Enclosed RWST (IRWST);
- Safety injection system
  - Four train direct vessel injection
  - Fluidic device in safety injection tank;
- Workstation based control room
  - Digital I&C
  - Hard wired backup in safety systems;
- Prevention and mitigation of severe accidents;
  - Cavity flooding system and in-vessel retention;
- Passive Auto-catalytic Recombiner (PAR).

3.3. Design optimization of the KNGR

With the completion of the basic KNGR design, we have decided to perform an integrated review on the design and to perform an optimization. The integrated review of the design was conducted from the perspective of the safety, economics, constructibility, operation and maintainability.

At the beginning of the optimization review, all issues have been collected and grouped. More than twenty items went through the optimization study.

Major items considered are: 1) Electric power up-rating, 2) NSSS and BOP safety system optimization, 3) Fuel and core design optimization for thermal margin and fuel performance, 4) Containment and severe accident mitigation system optimization, 5) General arrangement (GA) and building structure optimization for construction and maintenance convenience.

Table I shows the summary of the optimization evaluation and its determining factors related to safety, operational or cost impacts.

3.4. Advantages of the optimization

It was estimated that the removal of passive secondary condensing system (PSCS) and double containment is cost-effective. The cost benefit for the removal of double containment is more than 10 million dollars in direct cost savings without a large impact on safety. The final cost comparison after design optimization shows the cost is reduced by 60 million dollars per unit of original cost. The reduction in the construction duration is not credited at this cost assessment. The construction schedule experts estimated that the construction duration could be reduced by 1~3 months by elimination of outer containment.

TABLE I. OPTIMIZATION RESULTS BY EACH DESIGN ALTERNATIVES

Group	Items	Results	Remarks
Plant Power Level	-Electrical power up-rating (3,931→4,000MWth)	- 52" Last stage blade (LSB) adoption - Increase in fuel enrichment and new fuel in refueling	Cost saving 23M$/unit-year including 13M$ by power up-rating
NSSS Safety System	-Safety injection system with DVI -POSRV -Fluidic device in safety injection tank	- SIS with direct vessel injection - POSRV design - Fluidic device(FD) adoption	- No change from basic design
Fuel and Core Design	-24M fuel cycle -High burn-up fuel -MOX core design	- 18 Month fuel cycle - 30% MOX design cap.	- Change to 24 Month cycle if necessary - Long term R&D item
Containment and Severe Accident	-Double containment -Cavity flooding system(CFS) -Hydrogen mitigation system	- Single containment & in-vessel retention - Replacement of fusible plug with MOV(Motor Operated Valve) -Passive auto-catalytic recombiner + igniter	Accident mitigation Measure such as IVR adopted
General Arrangement	-Structural design optimization	- Compound building - System, building, structure optimization	Reduction of 5~10% of volume & bulk material
PSCS	-PSCS removal	- Removal of PSCS	Cost-benefit analysis
Performance Requirement	-Load follow capability -SG dry-out time	- Daily load follow - Relaxation of dry-out time to 20 minutes	-Excluding frequency control - Related to PSCS removal

4. DEVELOPMENT OF THE NUCLEAR PROJECT CONTROL SYSTEM (NPCS)

Recently, KEPCO has replaced the existing project management tool, ARTEMIS, with a newly developed computerized program, called NPCS, for construction of the nuclear power plants. The NPCS, developed by KEPCO, standardized all the construction working processes and introduced new information technologies (GUI, WEB) for user-friendly approaches.

The NPCS includes an abundant and accurate database, which could improve quality and efficiency of the project control.

Functions of the NPCS are to:

– Integrate schedules and actual progress analysis
– Control and forecast budget and actual cost
– Control material status
– Control drawings and documents
– Integrate information
– Create reports

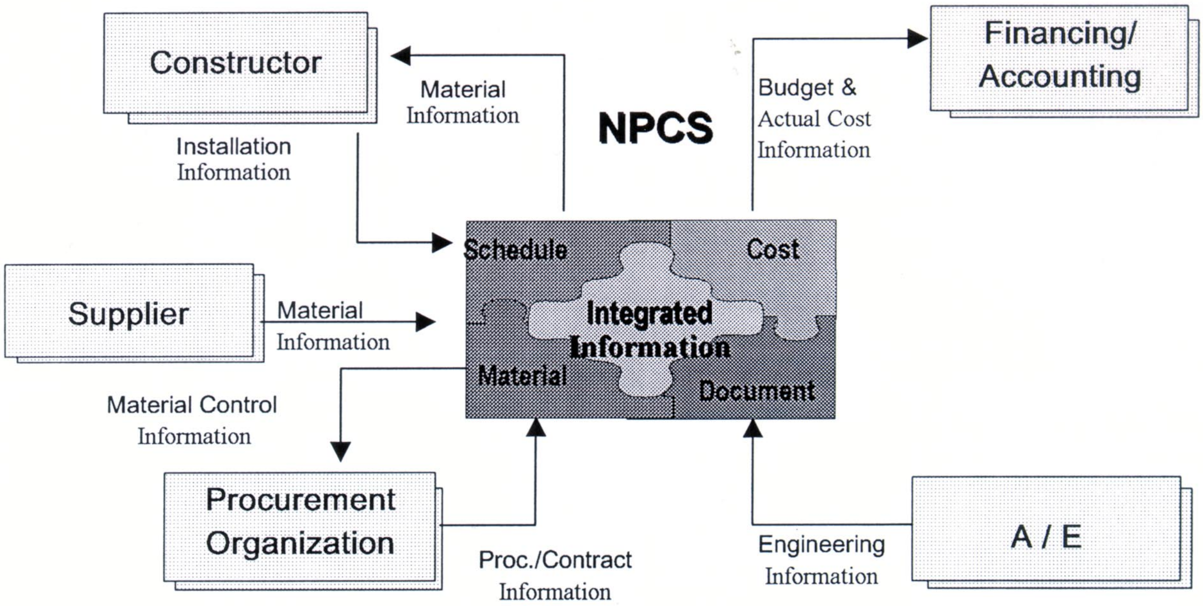

Fig. 2. NPCS structure

Figure 2 shows the structure of the NPCS and inter-relation among the organizations.

5. CONCLUSION

The mission of the KSNP$^+$ design improvement program and KNGR optimization program are to review and evaluate the optimization of plant arrangement, system design and the adoption of new system and construction technology. These programs are significant accomplishments in the continuous improvement in operability, maintainability, cost reduction, plant safety and establish the foundation of the marketable nuclear power plant concepts.

The successful construction and operation of the KSNP$^+$ and KNGR will definitely help KEPCO to form a firm foundation, to allow it to step forward into the advanced nuclear power technology era. We are looking forward to continuing design improvements and playing a central role in the future construction of safer and more economical international nuclear power plants.

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Cost reduction and safety design features of ABWR-II

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Abstract. The ABWR-II, which is aimed to be the next generation reactor following the latest BWR: Advanced Boiling Reactor (ABWR), is now under development jointly by the Japanese BWR utilities, General Electric Company, Hitachi Limited, and Toshiba Corporation. The key objectives of ABWR-II development include improvement in economics and further sophistication in safety for commercialization in the late 2010's and after. This paper summarizes the current status of ABWR-II development focusing on economics and safety. Plant power rating, fuel size, CRD rationalization and outage period are discussed from a cost reduction perspective. In terms of safety, the features such as diversification in emergency power sources and passive system application against severe accidents are being introduced.

1. INTRODUCTION

1.1. Background

The original Advanced Boiling Water Reactor (ABWR) was realized in the Kashiwazaki-Kariwa unit 6 & 7 after 20 years of development efforts since 1970. Ten more ABWRs are now under construction or planned in Japan [1]. In 1991, as soon as the first ABWR started its construction in Kashiwazaki-Kariwa in Japan, a new development programme for "ABWR-II" aiming to further improve and evolve ABWR was commenced. This early commencement was decided with considerations of long development period required for new generation reactors and on maintaining momentum and skill of technological development in the nuclear industry. The six Japanese BWR utilities led by Tokyo Electric Power Company and three BWR plant makers, namely General Electric Company, Hitachi Ltd. and Toshiba Corporation, have been jointly implementing the programme. This cooperative organization is basically the same as in ABWR development. The programme so far consists of three phases as shown in FIG. 1.

In Phase I (1991-92), future technologies were discussed and several plant concepts were studied. In Phase II (1993-95), in order to establish a reference reactor concept, key design features were selected. In Phase III (1996-2000), based on the reference reactor concept, modifications and improvements are being made to satisfy the design goals of ABWR-II, which are described later.

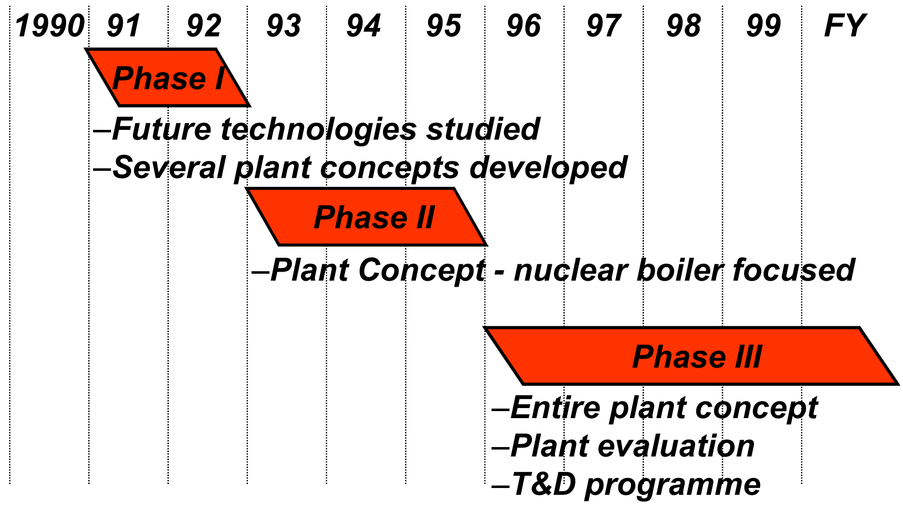

FIG. 1. ABWR-II development programme phases.

The commercial introduction of ABWR-II is now set for the late 2010's when replacements of old nuclear power plants are expected to start. Efforts are being made to render ABWR-II competitive in cost and to obtain public confidence on safety.

1.2. Design goals

The design goals for ABWR-II have been discussed since the beginning of this programme considering social and economic environmental changes along the way. For example, we encountered referenda and political decisions against nuclear power stations and deregulation in power generation industries.

The design goals are summarized as follows [2]:

(1) Enhancement in reliability and safety
(2) Reduction in human burden on operation and maintenance
 -Simplified design
 -Friendly-to-human man-machine interface
 -Better working environment
(3) Economic competitiveness against alternative forms of generation
 -Power generation cost reduction by increased availability
 -Capital cost reduction
(4) Flexibility for fuel cycle uncertainty

1.3. Design features

The current reference design concept shown in FIG. 2 includes the following design features:

- Large electric power rating of 1700 Mwe,
- Large fuel bundle of 1.5 times current BWR fuel bundle size,
- Rationalized CR/CRD by function,
- Large capacity SRV,
- Low pressure drop MSIV,

- Rationalized four division RHR,
- Diversified emergency power supply,
- RCIC with a generator,
- Passive heat removal systems.

These features will be explained in the following chapters in conjunction with their economics and safety.

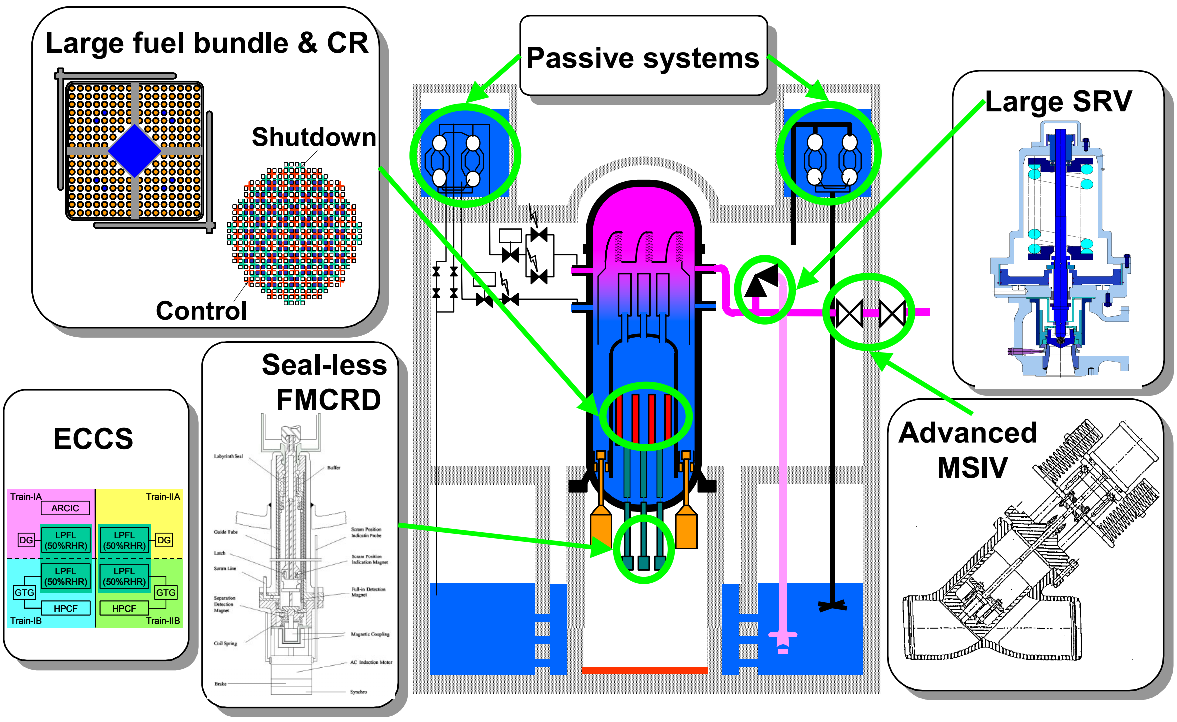

FIG. 2. ABWR-II design features.

2. ECONOMICS

2.1. Cost reduction target

In planning of future reactor, it is indispensable to set a cost target of power generation. It has become tougher and tougher 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 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.

2.2. Plant power output rating

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 30 % 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.

2.3. Core design improvements

Fuel bundle size increase of 1.5 times the current ABWR has been decided for the ABWR-II reference core design. The purposes of fuel bundle size increase are:

- To increase the space for fuel rods by reducing water gap area
- To reduce the number of fuel bundles.

The merits of having larger fuel bundles are:

- Higher core power density resulting in smaller RPV for cost improvement
- Increased flexibility to future fuel cycles including higher burn-up, MOX and higher conversion for optimized fuel cycle cost

In increasing the size of fuel bundles, shutdown margin was a key parameter. If the fuel bundle and control rod (CR) blade sizes are simply increased proportionally in the same arrangement, the shutdown margin will decrease. The current ABWR core is arranged in such a way that one cruciform CR is located in the center of every four fuel bundles, which is called a C-lattice arrangement. In order to avoid shutdown margin deterioration with a larger fuel bundle, it was decided to increase CR intensity, that is the number of CRs per fuel bundle, by having a K-lattice core arrangement. In the K-lattice core arrangement CRs are arranged at two diagonal corners of each fuel bundle to have CRs inserted on all four sides of each fuel bundle, which increases CR intensity to twice from that of the C-lattice as shown in FIG. 3.

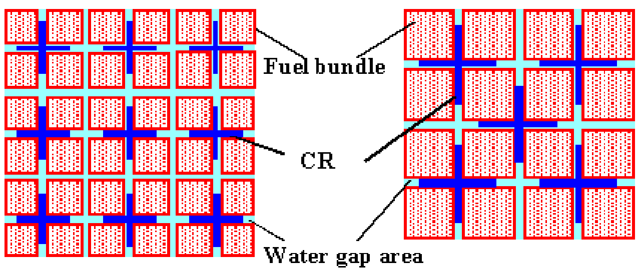

FIG. 3. C-lattice and K-lattice core arrangements comparison [2].

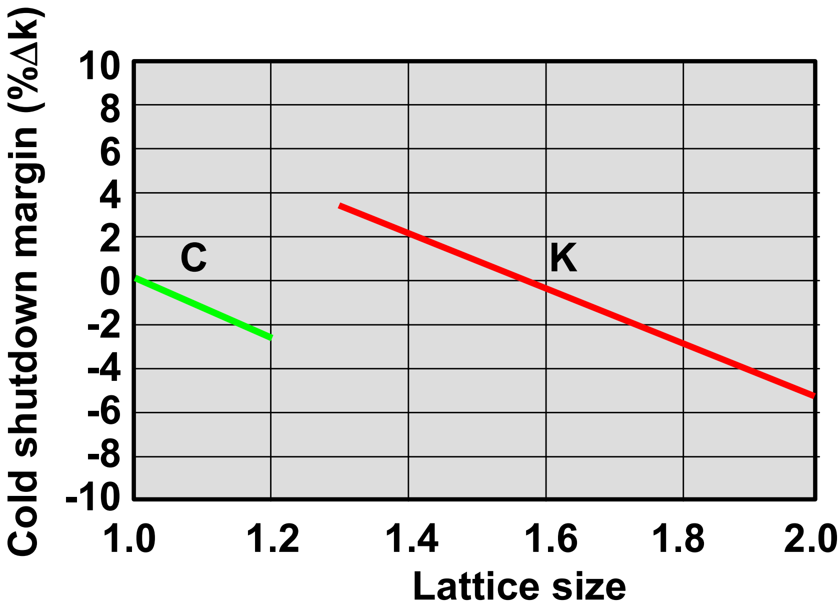

FIG. 4. Shutdown margin vs. bundle size in C and K lattice arrangement.

The shutdown margin in terms of fuel bundle size for C-lattice and K-lattice core arrangements is shown in Fig. 4. This figure shows that the current C-lattice shutdown margin can be maintained with a 1.5 times larger fuel bundle in a K-lattice arrangement.

The reference core design of the 1700 MWe ABWR-II is shown in TABLE I in comparison with the 1350 MWe ABWR and a conceptually extended 1700 MWe ABWR. The extended 1700 MWe ABWR core design is a simple proportional extension of the 1350 MWe ABWR core.

TABLE I. COMPARISON OF CORE DESIGNS FOR ABWR AND ABWR-II

	1350 MWe ABWR	1700 MWe Extended ABWR	1700 MWe ABWR-II
Core power density (kW/*l*)	50.6	50.6	58.1
Fuel bundle pitch (mm)	155	155	233
Number of fuel bundles	872	1100	424
Core diameter (m)	5.35	6.09	5.67
Number of CRs	205	269	197

The ABWR-II core has 15 % larger power density than the ABWR core. Therefore, the 1700 MWe ABWR-II core circumferential diameter is 0.4 m smaller than the extended ABWR's. This reduces RPV cost and facilitates compact arrangement of PCV.

In terms of the number of fuel bundles, the ABWR-II design has 424 bundles, which is approximately 60 % less than the extended ABWR. Therefore, a 60 % reduction in refueling time can be expected for ABWR-II.

As to the number of CRs, the ABWR-II core has 197 while extended ABWR with 1700 MWe output would need more than 30 % additional CRs. This will contribute cost reduction for the control rod drive (CRD) system in addition to the functional rationalization of CRDs that is described in 2.4.1.

From the above, the large (1.5 times) K-lattice core arrangement is expected to enhance the merit of scales of economy. As to operating cycle, an 18-months operation is selected as a reference based on a current optimization study result. Furthermore, the idea of spectrum shift rod (SSR) fuel bundle is now studied as an option to achieve efficient use of plutonium by water level control of SSR and rated power operation with all CRs withdrawn during most of the operating cycles. In this case, it is expected that fuel cycle cost will be reduced and that CRs need not be replaced periodically, which will contribute to additional refueling time reduction.

2.4. System and component optimization

2.4.1. CR/CRD system rationalization

With the ABWR-II 1.5 times K-lattice core, calculations show that insertion of half of the CRs is sufficient to achieve sub-criticality of the core when coolant temperature is close to the rated (hot shutdown). This means that the other half of CRs only have to be inserted by the time coolant reaches a cold state. So, rationalization of CR/CRD system in which only half of CRDs have scram function was considered. It will result in capital and maintenance cost reduction by eliminating hydraulic scram units for half of the CRDs. Therefore, the reference CRD design includes two kinds of functions and structures (See Fig. 5).

- Reactivity control CRD: Used for power control without scram function;
- Shutdown CRD: Used for hot shutdown with scram function.

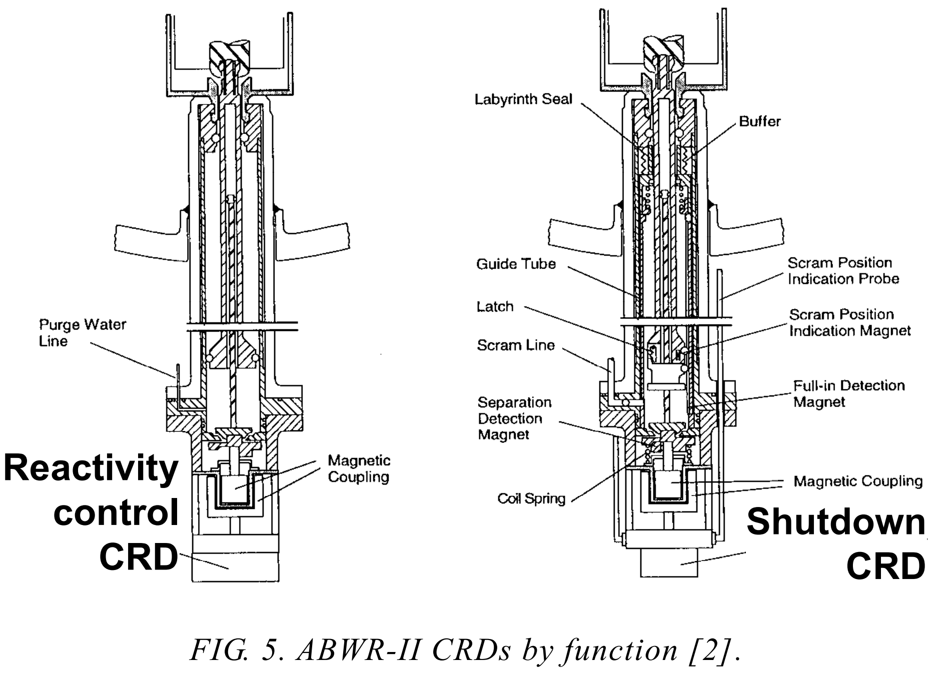

FIG. 5. ABWR-II CRDs by function [2].

In addition, 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.

2.4.2. RHR system optimization

The ABWR-II ECCS configuration is shown in Fig. 6.

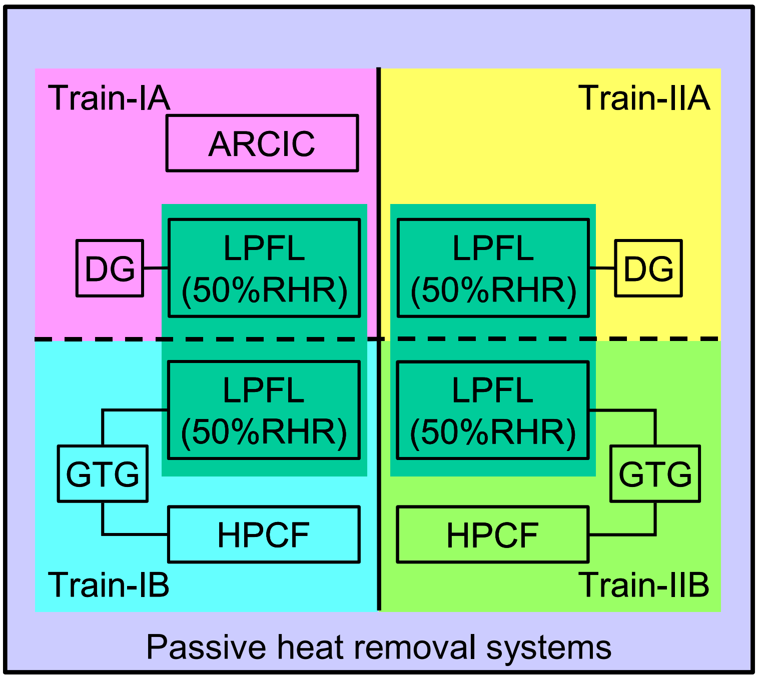

FIG. 6. ABWR-II ECCS configuration.

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.

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.

2.4.3. Main steam system component improvements

In dealing with the increased steam generation rate due to a larger thermal power, improvements from ABWR are made on two components: main steam isolation valves (MSIVs) and safety relief valves (SRVs). The independent development test programmes for these two components are proceeding with valve manufactures.

The bore diameter of the MSIV is increased and the center of gravity of its driving mechanism is lowered. (See Fig 7). This bore 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 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.

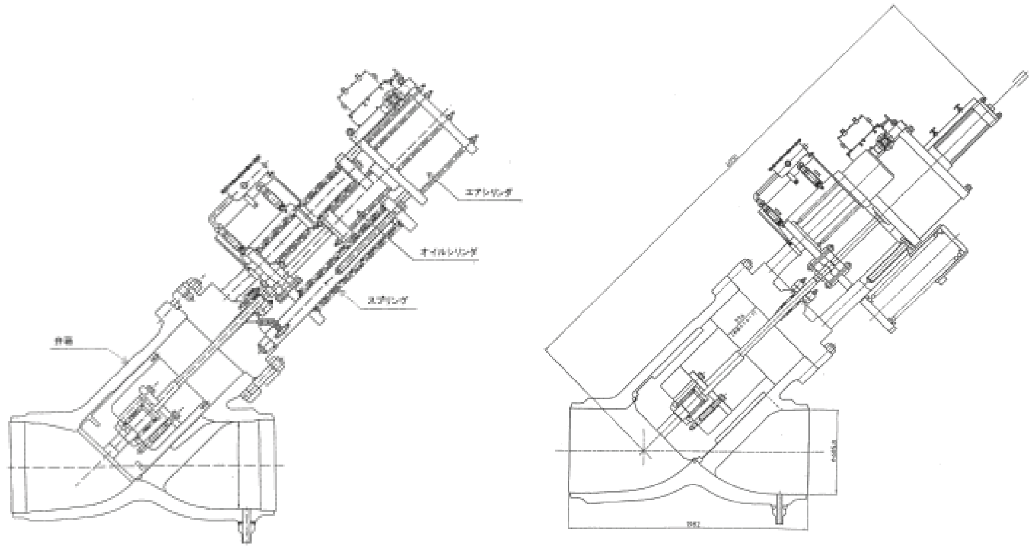

FIG. 7. ABWR (left) and ABWR-II (right) MSIVs comparison [2].

For the SRV, an increase of discharge capacity and simplification of valve structure were considered. (See Fig. 8). In order to minimize the number of SRVs, the discharge capacity per SRV is increased by 70 % to 680 t/h from ABWR's 395 t/h with an increased throat diameter and increased coil spring diameter. At the same time, the structure of the SRV is simplified by integrating an air cylinder into the SRV's main body. These improvements can reduce the number of SRVs for the ABWR-II from 23 to 14 and reduce the required number of SRV component parts, resulting in reducing capital cost and maintenance cost as well as facilitating layout of SRVs and their discharge system equipment.

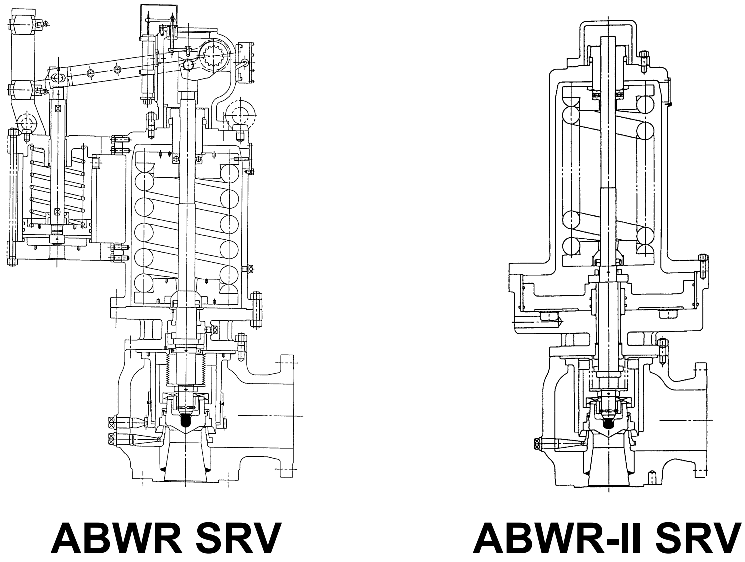

FIG. 8. ABWR (left) and ABWR-II (right) SRVs comparison.

2.5. Layout and BOP

Various PCV configuration and reactor building layout designs are being studied. The ABWR-II target is to provide a design that will minimize the cost increase from the 1350 MWe ABWR while accommodating a power increase and application of new passive systems.

The main turbine system is the same as the ABWR, that is a six flow turbine with 52 inch final bucket. It was confirmed that this turbine can handle 1700 MWe power generation. However, investigation of the turbine system was made looking for technologies that can contribute to compact arrangement for improved economy. The candidates for ABWR-II application are:

- In-line moisture separator and re-heaters;
- Feedwater pumps with high efficient fluid torque converter;
- Vertical feedwater heaters.

It was also confirmed that 1700 MWe generators can be manufactured by current technology.

2.6. Refueling outage period reduction

Refueling outage period reduction is a major factor in power generation cost reduction because it improves plant availability. 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 plant availability 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 plant availability through the plant lifetime would be expected to be more than 90 %.

2.7. Current economical estimation

Based on the current reference design concept, the plant capital cost for a 1700 MWe ABWR-II is estimated to be 102 % of that of the 1350 MWe ABWR, as shown below:

- Nuclear Boiler: +0.5 % of ABWR plant capital cost
- ECCS & Safeguards: +0.5 % of ABWR plant capital cost
- Turbine equipment: +2.5 % of ABWR plant capital cost
- Electrical equipment: +0.6 % of ABWR plant capital cost
- Buildings & structures: - 0.4 % of ABWR plant capital cost
- Construction period reduction: - 1.7 % of ABWR plant capital cost

Therefore, an approximately 20 % of specific capital cost (yen/kWe) is estimated.

Using the above capital cost, the power generation cost for ABWR-II was estimated with the following assumptions:

- Weighting factors: 65 % for capital; 20 % for O&M; and 15 % for fuel cost
- Availability: 93 % (7 % increase from ABWR's)
- O&M cost: 90 % of ABWR's
- Fuel cost: 100 % of ABWR's

ABWR-II power generation cost is estimated to be 77 % of ABWR's based on the above. More than a 20 % power generation cost reduction is expected by the current ABWR-II design. Further economic improvements are pursued toward the target of 30 % cost reduction.

3. SAFETY

3.1. Target on safety performance

The following safety related requirements have been established during early phases of ABWR-II development.

- 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 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 subsequent sections.

3.2. Design basis event

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 9 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.

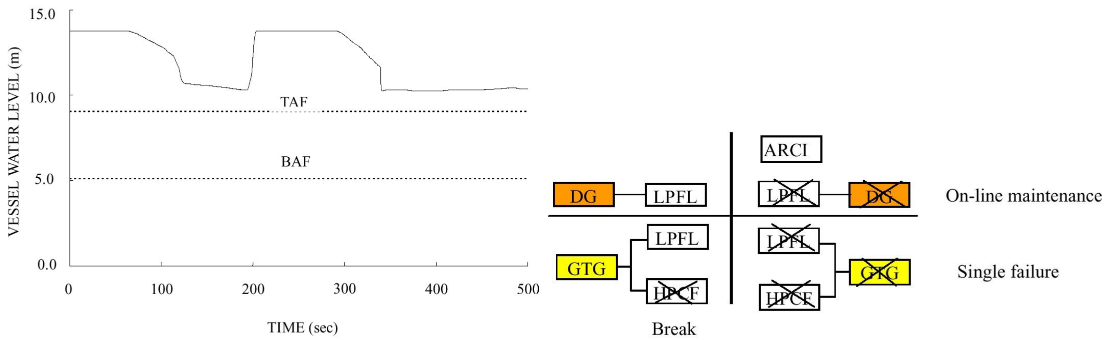

FIG. 9. The result of the DBA LOCA analysis by SAFER code [3].

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).

3.3. Beyond design basis event

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 system. 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 (Fig. 10). 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.

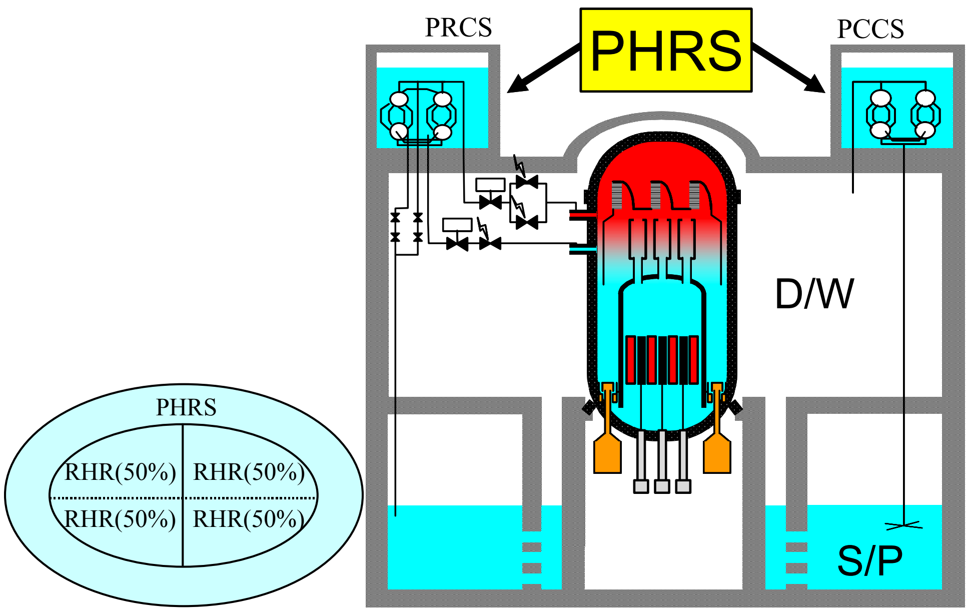

FIG. 10. Passive heat removal system.

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.

3.4. Current evaluation of safety performance

Although the major design effort has been focused on external/shutdown events as described in section 3.1, preliminary PSA evaluation shows that core damage frequency (CDF) for internal events during power operation has been reduced about one order (See Fig. 11) as a result of emergency power diversity and redundancy enhancement, passive cooling system installation, and RHR train redundancy enhancement.

Simplified PSA evaluation for design selection also provided the features of ABWR-II safety system configuration that is robust even in seismic induced events or shutdown events. Figure 12 shows a scoping result with simplified treatment of seismic event. Due to emergency power enhancement and dedicated passive cooling system, SBO sequence remains a small contributor even considering seismic induced events.

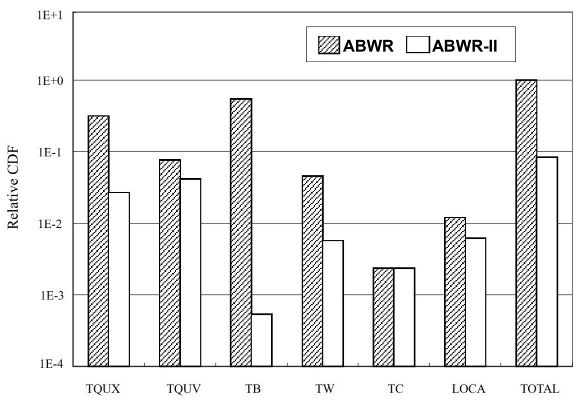

FIG. 11. Results of level 1 PSA for the ABWR-II and the ABWR [3]

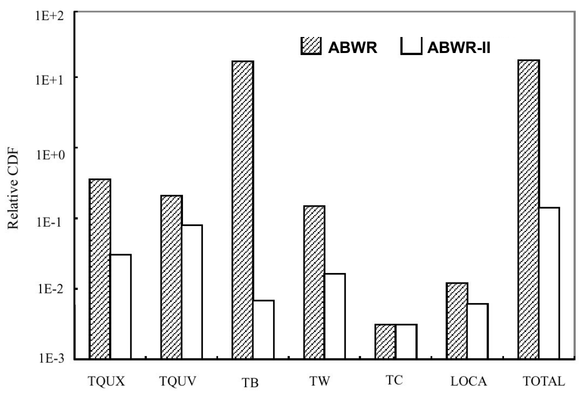

FIG. 12. Results of simplified seismic PSA for the ABWR-II and the ABWR [3]

When a shorter outage period is pursued to achieve higher plant availability, the core damage risk during the shutdown period might become a potential issue since the available number of RHR trains decreases almost throughout the outage period. The RHR configuration described in section 2.4.2 also contributes to reduce this shutdown risk.

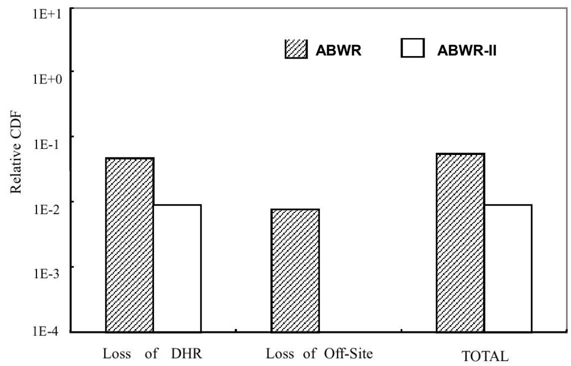

FIG. 13. Results of shutdown PSA for the ABWR-II and the ABWR [3].

As can be seen from Fig. 13 together with FIG. 11, shutdown risk is maintained to be about one order smaller than CDF during power operation, since RHR availability is kept even in the course of shortened annual outage periods by performing on-line maintenance.

4. CONCLUSIONS

The ABWR-II is now under development for improvement of economy and further sophistication on safety. The current reference concept is considered to have attractiveness in terms of capital cost, power generation cost and safety performance. However, efforts will be continued to render ABWR-II more attractive. In addition, on the technologies newly applied to ABWR-II, various testing programmes are performed or planned to consolidate their feasibilities and to find further room for improvements.

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Economical opportunities on advanced conventional island design for the European pressurized water reactor (EPR) based on Konvoi design

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Abstract. design of the European Pressurized Water Reactor (EPR) has been finalized by the end of 1998. In parallel with these efforts, the German utilities group contracted the Siemens AG Power generation Group (KWU) to develop an advanced and optimized conventional island for the EPR. The main objectives for improving the conventional island design were determined on the basis of experience of the Konvoi series plants and advanced fossil plants. This paper describes the innovations introduced to the conventional island and presents the reasons for the resultant cost reductions.

1. SUMMARY

German and French designers agreed in 1989 to jointly develop a standardized nuclear island for the European Pressurized Water Reactor (EPR).The basic design supported by German and French utilities and safety authorities was started 1995 and was finalized by the end of 1998. In parallel with these efforts, the German utilities group contracted the Siemens AG Power Generation Group (KWU) to develop an advanced and optimized conventional island for the EPR.

The main objective of the EPR design, i.e. to be able to compete economically with other nuclear power plant designs and fossil-fueled power plants and at the same time to increase nuclear safety, has been achieved. The results of these optimization efforts on the conventional island side can be summarized in the following points:

- The entire development and implementation process, i.e. from plant design work all the way through to plant service and maintenance, was reviewed and improved without any restricting operational or maintenance aspects;
- The efficiency of the steam, condensate and feedwater cycle, including the steam turbine and heat sink, was increased by introducing, among other design changes, the new 3DS/3DV blade design;
- The plant's electrical generating capacity was increased without any need of additional or new special tools or equipment;
- Common general European codes and related national codes and standards were applied to the designing, sizing, approval and documentation of all conventional island components;
- Only specialized personnel with global turn-key know-how was involved.

The performance figures of the improved design demonstrate the following:

- The EPR is economically competitive with modern fossil-fueled power plants,
- The EPR is much less dependent on fuel cycle costs than fossil-fueled power plants,
- One EPR saves some 10 million tons of CO_2 emissions per year compared with a hard-coal-fired power plant.

The result is a nuclear power plant with a gross electrical generating capacity of 1850 MW (for a site equipped with cooling tower), a gross efficiency rate of 37.8 % and a net efficiency of 35.9 %.

2. INTRODUCTION

2.1. Technical objectives of the EPR

In February 1995 Electricité de France and a group of German utilities (including PreussenElektra AG, Bayernwerk AG, RWE Energie AG, Badenwerk AG, Energie-Versorgung Schwaben AG, Isar-Amperwerke AG, Kernkraftwerke Lippe-Ems GmbH, Kernkraftwerk Stade GmbH and Neckar-werke Stuttgart), Framatome, Siemens AG Power Generation Group (KWU) and Framatome's and Siemens' joint venture company Nuclear Power International (NPI) signed a contract to develop the basic design of the nuclear island.

The result is the European Pressurized Water Reactor (EPR) with an evolutionary nuclear reactor design derived from the German Konvoi series and the French N4 series.

The main technical objectives of the European Pressurized Water Reactor project are

- To be able to compete economically with other nuclear power plant designs as well as hard-coal-fired power plants;
- To provide satisfactory performance characteristics, such as a generating capacity of about 1850 MWe, a plant service lifetime of 60 years for non-replaceable components, an average availability over the plant's lifetime of > 90 %, an average duration of scheduled refueling outages of ≤ 19 days per year and average inadvertent unavailability of < 5 days per year, as well as other objectives specific to the nuclear island;
- To fulfill German and French public power grid requirements;
- To increase safety by reducing the risk of accidents and mitigating the consequences of severe accidents by implementing accident control design features;
- To be licensable in Germany and France;
- To achieve a plant construction period of 57 months beginning with first concrete for the foundation raft.

2.2. Additional technical objectives of the conventional island

In the same year the German utilities involved contracted Siemens to develop an optimized conventional island design for the EPR in line with the same main technical objectives defined within the scope of the Franco-German cooperation agreement.

The following technical objectives for the conventional island were given highest priority with regard to improve efficiency, reliability and economy of the EPR plant:

– Excellent plant operational behavior and a design which allows flexible, unrestricted maintenance, i.e. as good as or better than the Konvoi series design;
– Mean forced outage time per year for the turbine-generator-set much smaller than 40 h per year;
– Risk-free manufacturing of components based on existing technology and proven design principles and, as far as possible, without need for additional or new special tools and equipment;
– Transportability of heavy components by railway using existing transport trolleys, e.g. for the generator and transformers;

As reference engineering and know-how
– the advanced design of Siemens' Konvoi series plants, with its excellent operational behavior and economy,
– publicly available data on the designs developed by other turbine generator set suppliers, and
– the experience gained from advanced fossil-fueled power plants

were agreed upon as bases for the optimization work.

The study conducted by Siemens is based on a river site with cooling tower, and was carried out in accordance with the standard site conditions for the nuclear island. All optimization measures described below are transferable to an EPR plant with direct cooling. Under Section 3 below, the main data for an EPR with direct cooling are compared to an EPR equipped with cooling tower.

3. MAIN RESULTS OF THE INNOVATIVE CONVENTIONAL ISLAND

The essential objectives for improving the conventional island with respect to lowering capital investment and power-generation costs were determined on the basis of the experience gained from the design, construction, erection, operation and maintenance of the Konvoi series plants and advanced fossil-fueled power plants.

These essential objectives are the following:

– Increased plant output;
– Increased turbine and steam, condensate and feedwater cycle efficiency and availability;
– Optimized and simplified mechanical and electrical systems and functions taking into consideration the high level of equipment quality;
– Applied common, general European codes to all conventional island components without reducing equipment and system reliability and availability;
– As far as reasonable, use of proven design principles, existing technology and familiar manufacturing procedures.Provision of physical separation of conventional island and nuclear island functions;
– Reduction of the enclosed volume of buildings without restricting erection, service or maintenance work;
– Simplification of instrumentation & control and mechanical systems without restricting operation by implementing advanced instrumentation & control systems;
– Same type of operational I & C as applied for the nuclear island.

The innovations introduced to the conventional island in comparison to the Konvoi series design are described in the following subsections.

3.1. Steam, condensate and feedwater cycle

The general architecture of the steam, condensate and feedwater cycle for the EPR is shown in Fig. 1, including one double-flow high-pressure turbine section, three double-flow low-pressure turbine sections, seven extraction stages, four feedwater pumps and one spray-type feedwater buffer tank.

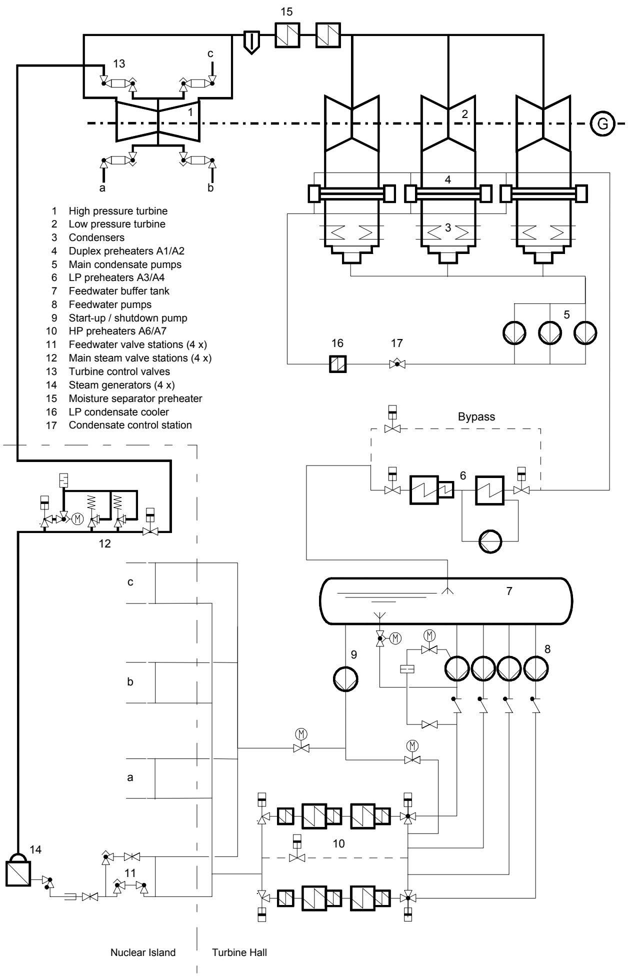


FIG. 1. Steam, condensate and feedwater cycle.

The major improvements to the steam, condensate and feedwater cycle over the Konvoi series design can be divided into the following two categories:

– *Innovations which have a direct impact on lowering investment costs*

These innovations comprise simplifications or reductions of systems or equipment which have no direct effect on plant availability or the application of systems and equipment which have an acceptably low failure rate.

These modifications were based entirely on the operational experience of the utilities and the designer, and resulted for example in no reduction in the number of main condensate pumps, with three 50-% capacity units, and the main feedwater pumps, which are configured as four 33-% capacity units instead of in a 3 x 50 % arrangement.

The following are examples of such innovations:

- Reduced number of low-pressure main condensate trains;
- A smaller feedwater (buffer) tank;
- Integration of cooler function into preheaters;
- Modified warmup procedure for the steam, condensate and feedwater cycle which has ultimately led to the elimination of an entire building, i.e. the auxiliary steam supply system building;
- Elimination of the condensate polishing system based on the good experience gained with steam, condensate and feedwater cycle chemistry together with improvement in material composition and component design.

– *Innovations which increase electrical power output and ultimately decrease specific power-generation costs*

A decrease in power-generation costs can only be achieved if any additional costs for equipment are lower than the gain in additional electrical power output.

Examples of such process improvements are:

- higher main steam pressure,
- higher final feedwater temperature,
- optimized feedwater (buffer) tank pressure,
- optimized number of steam turbine extraction stages,
- two-stage reheating in a vertical design as used in the Konvoi series design,
- optimized preheater efficiency,
- higher turbine efficiency together with an optimized heat sink (see details below).

The gain in electrical generating capacity in comparison with the Konvoi plants amounts to approximately 67 MW. This gain can be broken down as follows:

- 50 % of this increase or more is due to design improvements to the steam, condensate and feedwater cycle, while
- up to 50 % of this increase can be traced to improvements to the high- and low-pressure turbine sections, including the new blade design.

Some of these improvements have already been backfitted at Isar 2 Nuclear Power Station (one of the Konvoi series).

3.2. Steam turbine-generator set including heat sink

3.2.1. Steam turbine

Basis for the steam turbine plant is the proven, highly reliable product installed in the Konvoi series plants. In light of the experience gained from state-of-the-art turbines used at today's fossil-fueled power plants, certain modifications were introduced to achieve higher efficiency.

The main design features of the EPR Siemens steam turbine are as follows:

- 1 double-flow high-pressure turbine section and 3 double-flow low-pressure turbine sections;
- 2 instead of 4 inlet and outlet nozzles in the high-pressure turbine sections;
- High-pressure turbine section upper casing equipped with only one nozzle;
- Use of blades shaped using 3D design techniques for the high- and low-pressure turbine sections (see details below);
- Combined main steam control and stop valves arranged below the turbine floor instead of above the high-pressure turbine section;
- Electrohydraulic valve actuators with external high pressure control fluid supply (compact actuators were not used for reason of costs);
- Low-pressure turbine section outer casing directly welded to the condenser shell, which rests directly on the foundation without springs;
- Support for the low-pressure turbine section inner casing rests directly on the turbine foundation deck and not in the outer casing;
- With the exception of the final-stage moving blade rows, the low-pressure turbine section blades are designed with shrouds;
- Intercept butterfly valves are not required according to German VGB rules and international (IEC) requirements.

3.2.2. Steam turbine blading

Turbine blade design has a major influence on the quality of energy transfer. In earlier years, it was only possible to use cylindrical blades of uniform profile. Design improvements were made to these cylindrical blades time and time again. Increasing understanding of turbine processes and improved manufacturing machinery made possible improvements in efficiency.

Today, powerful computers make it possible to resolve the system of differential equations in greater detail using some 80 free parameters limited by approximately 300 constraints such as material characteristics as well as design, fabrication and erection requirements. Such methods provide better understanding of ideal blade shape. On the basis of this new technology it has become feasible to develop blade shapes optimized in three dimensions in which profile, twist and slope change over the entire length of the blade.

This approach permits better adaptation to the complex radial flow distribution and to the specific steam conditions between stationary and moving blades, and helps to reduce secondary losses at the root and tip of the blades. The resulting product has been dubbed a 3DS blade.

There remained further potential for improvement in terms of adapting blade geometry to the specific flow and steam conditions in the various turbine stages. The distribution of the pressure decay per stage over the stationary and moving blades - the so-called mean reaction - was more or less uniform.

Now, Siemens is the first blade manufacturer to succeed in adapting the blade shape for each stage separately. Called 3DV blades, these designs eliminate the classic distinction between impulse (with a reaction of about 0 %) and reaction turbines (which have a reaction of 50 %).

Figure 2. shows the new rotating blades with three-dimensional shapes (3DS and 3DV) in comparison with cylindrical blades.

FIG. 2. Cylindrical blade *3DS/3DV blade*

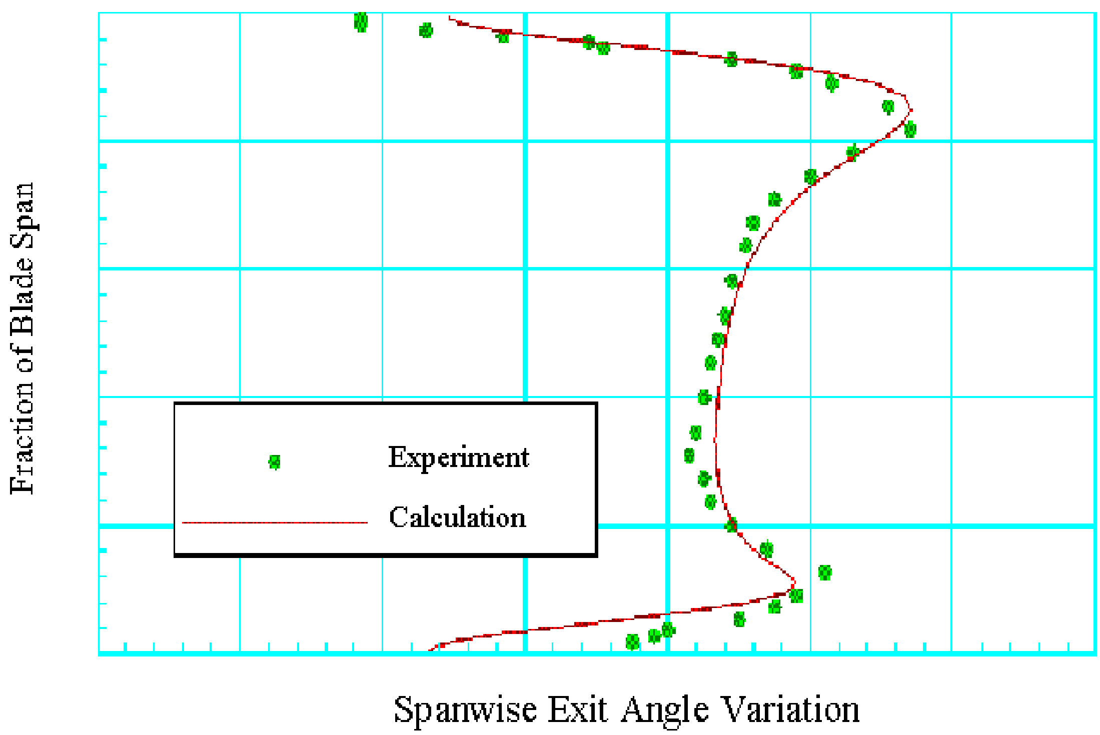


FIG. 3. Calculated and measured flow angles.

In comparison to cylindrical blades, these individual optimizations over the lengths of all the turbine blades and across the various stages result in a gain of 2 to 3 percentage points in efficiency in each turbine section and up to 1 percentage point in total for the overall plant.

At Siemens' manufacturing plant, improved computerization facilitates not only precise visualization of shape and strength analyses but also direct transfer of blade shapes to the fully automated fabrication cells. This results in lower blade failure rates, higher quality and lower cost.

The methods applied make possible quite accurate simulation of boundary and secondary flows in critical areas. Analytical results have been confirmed by numerous experimental measurements performed in wind tunnels and on test turbines.

The good agreement of calculated and measured flow angles for 3DV variable reaction blading is shown in Fig. 3.

3DV blades are shaped such that the thermodynamic energy of the steam is optimally converted into rotating shaft mechanical energy, with reduced secondary losses. These blades significantly improve steam turbine efficiency.

3.2.3. Generator

With the exception of the rotor cooling system, the EPR's generator system is based on the improved and highly reliable product used in the Konvoi series. The coolant for the rotor winding cooling system has been changed from water to hydrogen.

The main design features of the EPR generator are as follows:

– single-shaft turbine generator set
– 2056 MVA, power factor 0.9, 27 kV, 44 kA, 4-pole type
– hydrogen cooling system for rotor winding and stator core
– water cooling system for stator winding
– end shield bearings with integrated hydrogen sealing
– total weight: 899 Mg; total length: 22.4 m (including exciter set).

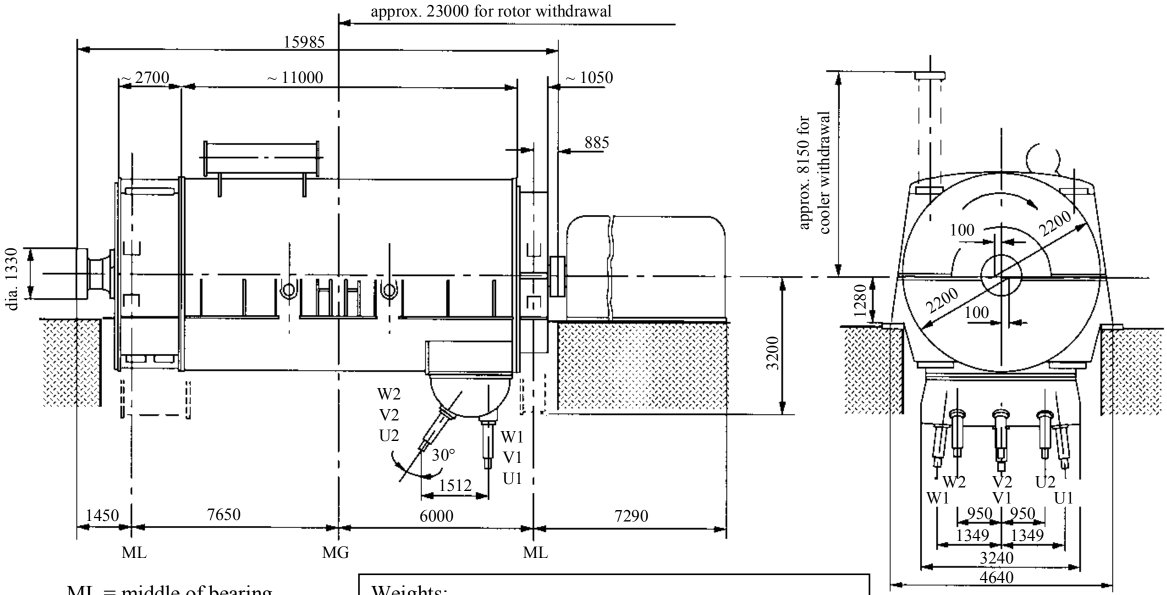
The design of this generator series, up to a power range of 2200 MVA with a power factor of 0.9, has produced an advanced product. It is still based on the identical design principles of proven generators with water- or gas-cooled rotor winding, and remains in compliance with the requirements defined by the International Electrotechnical Commission (IEC).

This advanced generator with gas-cooled rotor winding has the following advantages over an equivalent generator with water-cooled rotor winding:

– shorter total length
– less total weight
– fewer active components
– simplified cooling system
– simplified rotor bar fabrication
– simplified operation and maintenance work.

This advanced Siemens generator shown in Fig. 4 is not larger than the water-cooled generators installed in the Konvoi series, with an apparent power of 1640 MVA, and is consequently also transportable by road as well as by railway using an existing transport trolley. The weight of the heaviest generator component is still less than the maximum weight of 450 Mg allowed for railway transportation.

Siemens manufacturing facilities are already equipped with the machinery needed to fabricate this generator with hydrogen-cooled rotor winding, the world’s largest generator of its kind.



ML = middle of bearing
MG = middle of generator

Weights:	
Rotor	249 Mg
Stator complete (incl. cooler, bearing, housing)	521 Mg
Transport weight, stator	434 Mg
Cooler	9 Mg
Exciter	75 Mg
Others	54 Mg
Total weight	899 Mg

FIG. 4. EPR generator.

3.2.4. Heat sink

Optimization of the heat sink, which can have significant effect on generator output, auxiliary power load, component costs and plant layout, is one of the essential steps towards achieving high efficiency while keeping an eye on high plant economy. Site conditions for a cooling tower or direct cooling system, interface requirements to the nuclear island and fuel-cycle costs must be considered in addition to overall plant requirements.

The major input data for optimization work are the following:

- Air and cooling water temperatures for a site equipped with cooling tower, or the given temperature rise and circulating water temperature for a direct-cooling system,
- Electrical power output,
- Operating time per year at full-load power operation,
- Heat consumption,
- Fuel-cycle cost,
- Cost of specific plant systems and components taking into consideration plant layout constraints,
- Anticipated investment cost of the overall plant,
- Annuity.

The optimized design aspects which lower power-generation costs are:

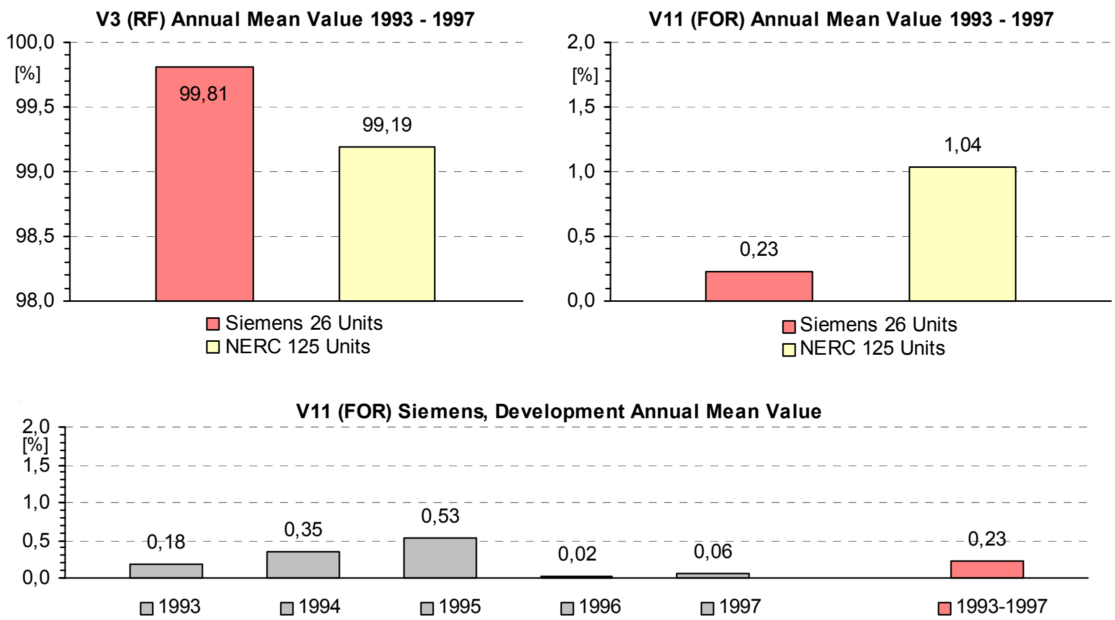
- Condenser pressure and length,
- Circulating water flow rate and velocity,
- Design of the cooling tower and circulating water structures.

One example of the EPR's optimized heat sink is the lower circulating water flow rate required, which is about 15 % less than flow at the Konvoi series plant Isar 2.

Furthermore, a significant additional increase in generating capacity, higher efficiency and lower power-generation costs can be achieved in cases in which site conditions allow direct cooling of exhaust steam rather than requiring a closed cooling system. A gain of 20 MW is realistically feasible given a cooling water temperature of between 12 and 16°C.

3.2.5. *Availability*

In Fig. 5 values of availability of Siemens turbine-generator-units are compared with values of NERC units (North American Electric Reliability Council).



$$\text{V3 [RF]} = \frac{\text{Period Hours} - \text{Forced Outage Hours}}{\text{Period Hours}} \times 100\ \% \qquad \text{V11 [FOR]} = \frac{\text{Forced Outage Hours}}{\text{Operating Hours} + \text{Forced Outage Hours}} \times 100\ \%$$

FIG. 5. Turbine-generator, availability.

The upper left diagram shows the availability based on period hours (= 8760 h/a) and the upper right diagram the forced outage rates based on operating hours. Evaluated years (1993 to 1997) are detailed in the next diagram.

The mean forced outage time in hours per year for the complete turbine-generator sets of Siemens and NERC-unites are as follows:

		period	1988 – 1992	1993 – 1997
□ Siemens	unit	hours	25	17
	turbine / generator	hours	13 / 12	10 / 7
□ NERC	unit	hours	70	71
	turbine / generator	hours	39 / 31	33 / 38

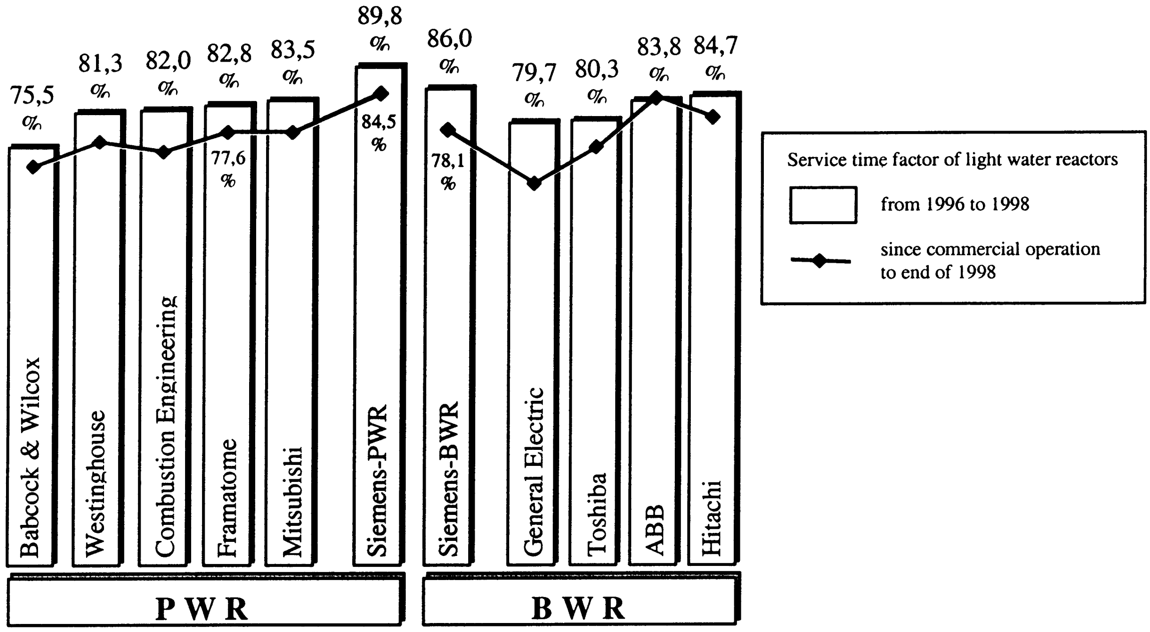


FIG .6. Service time factor of light water reactors (total plants).

Since Siemens did not change general design principles of the turbine and generator for the advanced and optimized conventional island for the EPR these really excellent values of Siemens units can be expected without any doubt also for the EPR plant. Beyond it, the availability of the total plant (NI and CI) also indicates quality just as economy. Fig. 6 shows the availability of Siemens plants (NI and CI) against international competitors.

Of course, the figures show an improvement of availability since commercial operation for pressurized water reactors (PWR) and boiling water reactor (BWR) of all suppliers, but Siemens plants fill the first places.

The higher availability of Siemens plants from 1996 to 1998 (at least 6.3 % for PWR and 1.3 % for BWR) and also since commercial operation demonstrates their quality and at the same time their economy of operation.

3.3. Electrical system

The main structure of the electrical system for the conventional island is similar to that of the Konvoi series, and follows the 4-train redundancy requirement specified for the nuclear island.

Fig. 7 shows the architecture of the electrical system of the conventional island.
The power plant is connected to two off-site power grid systems: one main grid connection for power transmission and normal plant startup and shutdown, and one independent auxiliary grid connection for plant shutdown in the event of simultaneous loss of the main grid connection and main generator.

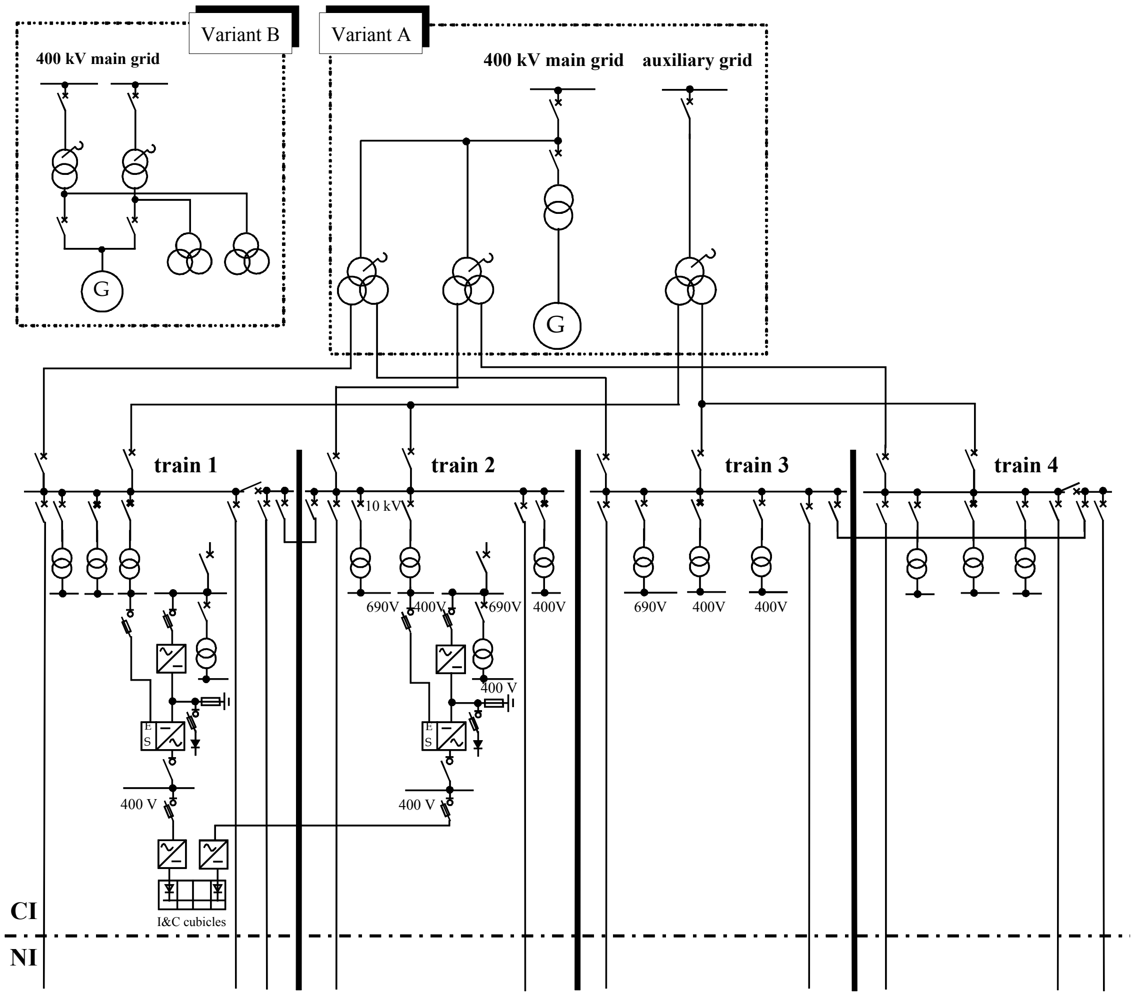


FIG. 7. Single line diagram of the conventional island.

Additionally, some consumers of the conventional island are connected to the emergency power supply of the nuclear island for investment protection (such as the turbine) and for discharge reduction of the batteries in case of loss of the above mentioned two offsite power grid systems.

The results of optimization of the electrical system over the Konvoi series design can be divided into the following three categories:

- Modifications with interface to the public power grid,
- Modifications with interface to the power supply of the nuclear island,
- Modifications which lower investment costs.

All of these modifications fulfill both French and German requirements despite the differences in general grid structures such as load concentration, power transmission distance, load flow, network node for power control and safety rules & regulations, as well as requirements governing power supply of the nuclear island via the auxiliary or standby transformers.

The following are examples of improvements to the electrical system:

- Three 400-kV single-phase main transformers with build-on air/oil coolers instead of two 3-phase transformers with separate water/oil coolers;
- Indoor 400-kV generator circuit breakers instead of 27-kV units;
- Two 400-kV/10-kV 3-phase auxiliary transformer and one 400-kV (or 220-kV or 110-kV) standby transformer;
- Physical separation of conventional island switchgear from the nuclear island;
- Dry-type low-voltage transformers;
- Intelligent switchgear and intelligent drive actuators, including bus systems;
- 230V AC power supply for instrumentation & control equipment.

These modifications bring the following benefits:

- Easier manufacturing and transportation of main transformers by railway or road,
- Simpler erection and removal of main transformers,
- No cooling water system connection required for main transformers,
- Simplified generator leads,
- Identical auxiliary and standby transformers,
- Shorter cable runs from auxiliary transformers to the switchgear and loads,
- Less cabling between I&C cabinets and switchgear,
- Continuous monitoring of equipment,
- No DC switchgear.

These improvements lower investment costs, ultimately decrease power-generation costs and have no adverse effect on plant availability.

3.4. Instrumentation and control (I & C)

In order to improve the economy of the overall plant, the instrumentation & control (I&C) systems, too, required optimization, taking into consideration requirements governing the nuclear island as well as operational & safety plant behavior and performance.

First, the architecture of the operational I & C of the nuclear and conventional islands was harmonized with a view of achieving a homogeneous man-machine interface. The cost reduction potential of the conventional island was determined and evaluated. The optimizations identified for implementation were closely matched to nuclear island requirements.

Figure 8 presents an overview of the operational and the safety I & C systems for the entire plant (NI and CI). Process variables are monitored by the operational and safety instrumentation. Many of the measured values generated by the safety I & C are also used for normal operation.

The introduced main improvements of the conventional I & C systems based on a homogeneous architecture are:

- Comprehensive use of a fiberoptic network;
- Use of standardized instrumentation, branch connections and automation systems;
- Standardized blackbox controls and interfaces;

– Use of field bus systems together with intelligent switchgear and actuators;
– Direct supply of 230V AC power to I&C equipment cabinets using integrated 230V AC/24V DC converters and, as far as reasonable for specific equipment, without power conversion;
– Optimization of equipment arrangements.

These improvements result in unproblematic signal transmission over long distances, inherent resistance to electromagnetic interference, provisions for physical separation, fewer I & C equipment cabinets and branch connections, less cabling and auxiliary power requirements as well as simpler design, assembly, erection, commissioning, maintenance and documentation. Consequently, equipment costs are lower.

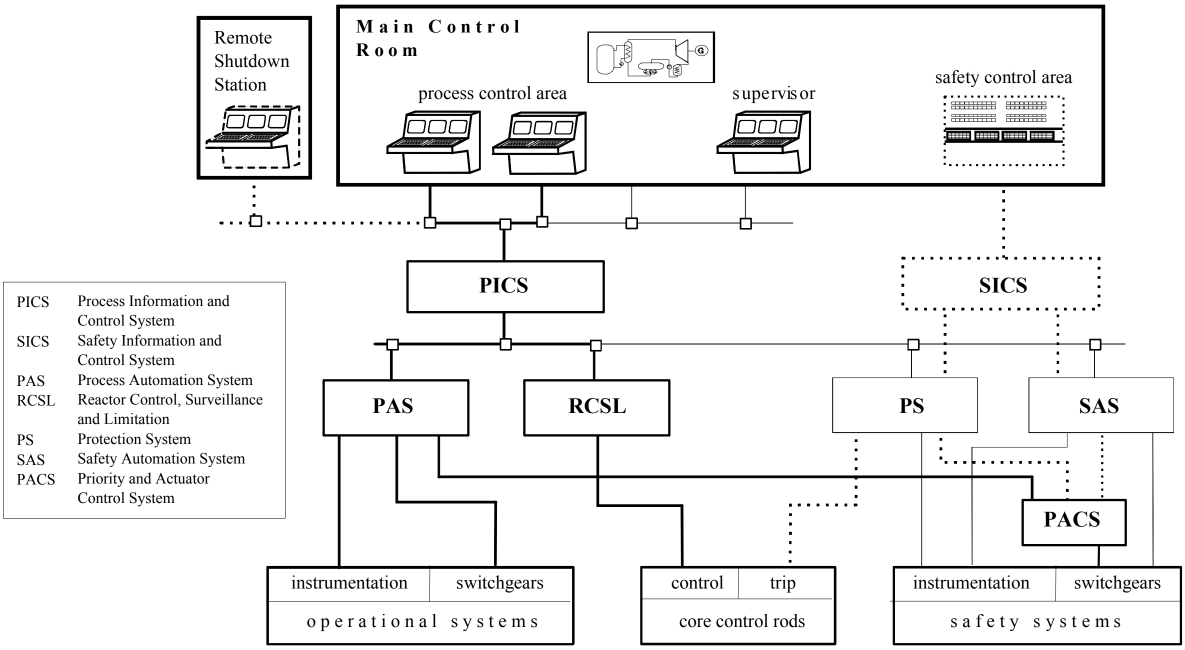


FIG. 8. Architecture of I & C systems.

3.5. Components specifications

The manufacturing and quality specifications defined for the Konvoi series covered the conventional and nuclear islands. These specifications stipulate requirements for design, sizing, materials, manufacturing and the preparation of pre-approval documents as well as for documentation of nuclear island and conventional island components. The philosophy behind this concept was to achieve a compact and homogeneous quality code for the complete nuclear power plant.

With this concept in place, however, it was very difficult for foreign suppliers and sub-suppliers to gain access to the German nuclear power market due to the specific quality certificates required. Furthermore, the small group of suppliers qualified to these requirements enjoyed a virtual monopoly and took advantage of this situation.

In future, only common, general European codes and the related national codes and standards will be applied to the EPR conventional island components. This ensures an equivalent quality of components fabricated in all member nations of the European Union.

The advantages of this new concept include the following:

- All European manufacturers will have a fair chance in competition since all national requirements for design, sizing, approval and documentation will be based on common European codes;
- All European manufacturers are familiar with the national codes applicable in their respective countries;
- Manufacturers need not implement any special, additional quality management program;
- The manufacturers' qualifications and product quality of the European suppliers will be comparable since qualification procedures, like the training of skilled workers and experts, are comparable;
- Costs will be as low as those for fossil-fueled power plants.

These advantages produce benefits in the procurement of components which meet high quality management requirements, resulting in lower investment costs and ultimately lower power-generation costs.

3.6. Plant layout

3.6.1. Site plan of the EPR with cooling tower (Single-Unit Arrangement)

The turbine building is arranged in axial line with the reactor and safeguard buildings. On one side of the turbine building, the circulating water system is arranged at a right-angle to the turbine building, while the conventional island switchgear building and auxiliary buildings (workshop, office and entrance buildings) are located on the other side.

The main reasons for this arrangement are the following:

- Safety requirements specified for the nuclear island buildings and associated equipment;
- Economic flow of fluid and energy between the buildings and equipment;
- Economic arrangement of circulating water system and power grid connection;
- Economic requirements governing civil construction, component erection and plant operation.

Modification of circulating water piping and channels for a plant site with direct cooling (river or seaside) or for a twin-unit arrangement can be easily implemented.

Fig. 9 shows the overall arrangement of a single-unit EPR for a site with cooling tower.

3.6.2. Turbine building

The layout of the conventional island building has been totally rearranged due to the results of optimization of systems, components and equipment.

The main requirements governing this rearrangement are the following:

- Reduction of enclosed volume of buildings;
- Adaptation of layout to accommodate optimized mechanical and electrical systems;
- No changes shall be made to the vertical arrangement of the moisture separators;

– Improvement of the service area for the combined main steam control and stop valves and the high-pressure turbine section;
– No reduction of area provided to perform maintenance work, e.g. for the steam turbine and pumps;
– Provision of physical separation of nuclear and conventional island electrical and I&C systems;
– Rearrangement of areas for laboratories, workshops and stores;
– Reduction of required number of operating personnel.

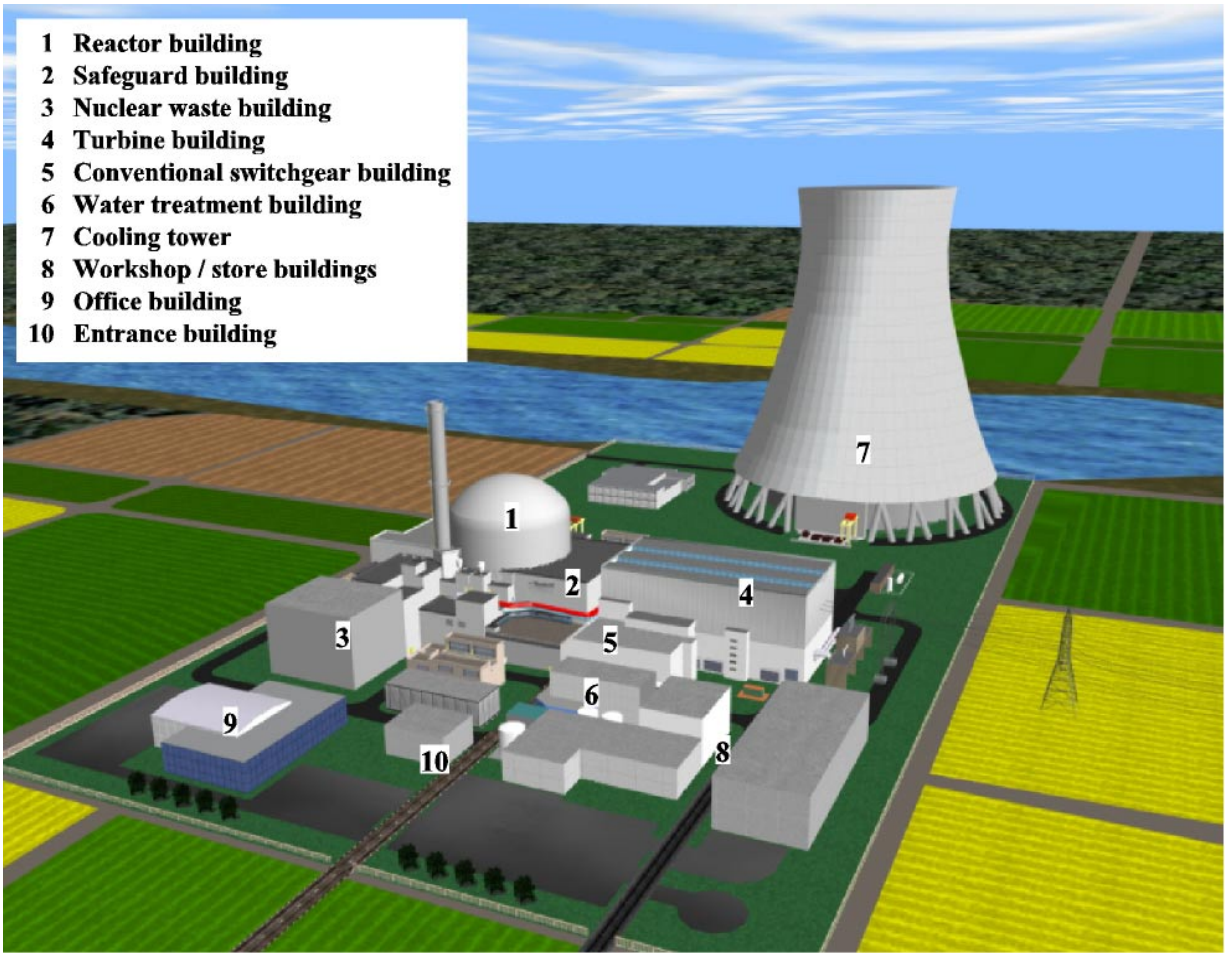


FIG. 9. 3D-arrangement of the EPR-plant.

Optimization work of the turbine building for the EPR with 1850 MW, viewed against previous layouts of nuclear power plants rated at around 1400 to 1500 MWe, reduced enclosed building volume by some 10 % compared to the Konvoi series and by some 50 % compared to competitor vendors. Areas for service and maintenance work were not reduced, thus ensuring short erection and maintenance times.

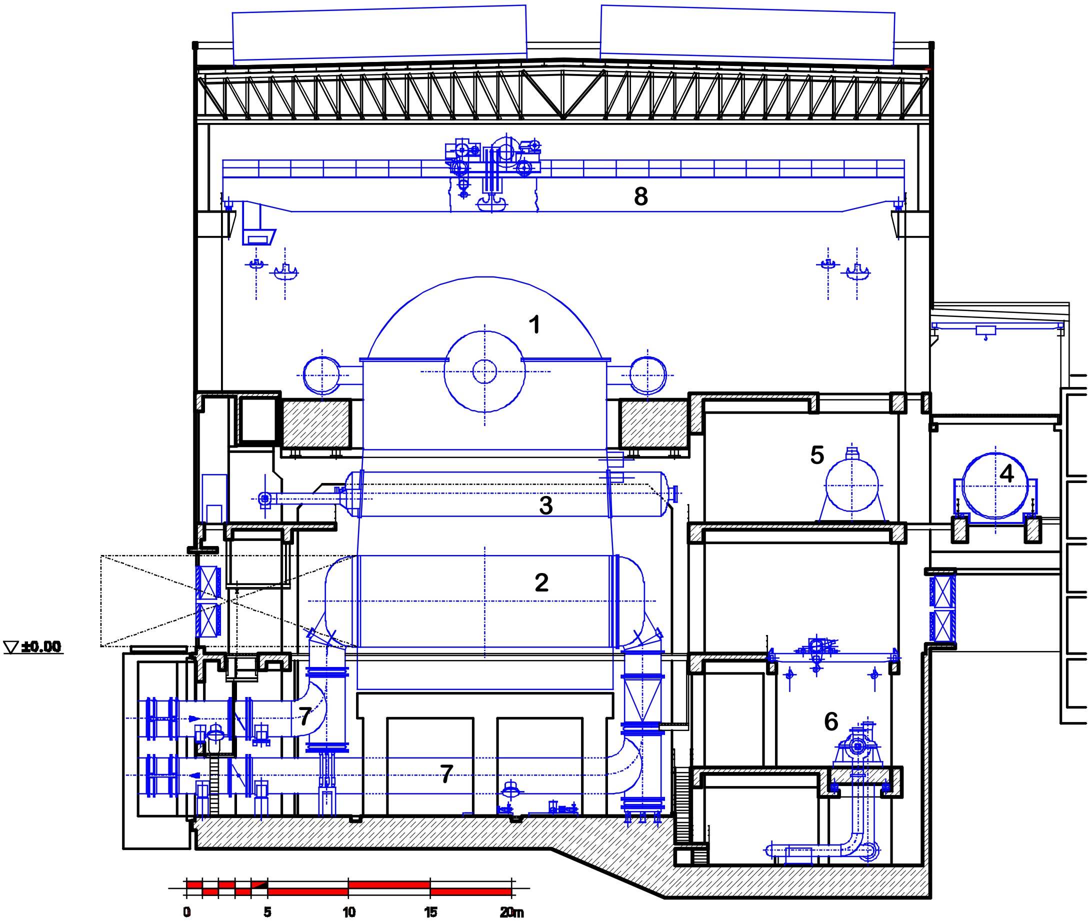
The following are examples of layout rearrangements introduced to the turbine building:

– Elimination of feedwater bay due to rearrangement of the feedwater pumps and optimized location of the feedwater buffer tank;
– Rearrangement of main steam stop and control valves;
– Rearrangement of building floor elevations;

– Rearrangement of preheaters;
– One instead of two high-pressure turbine section inlet and outlet lines;
– Simplification due to compact turbine auxiliary control and lubrication systems;
– No need for butterfly intercept valves.

This work involved an integrated team with good overall engineering knowledge of the civil works, systems and components, I&C and electrical equipment as well as know-how in the fields of civil construction, component erection, plant commissioning and inservice inspection and maintenance work.

Figure 10 shows a cross section of the turbine building.



1	**LP Turbine**	**5**	**LP Preheater**
2	**Condenser**	**6**	**Feedwater Pump**
3	**Duplex Preheater**	**7**	**Circulating Water Pipes**
4	**Feedwater Buffer Tank**	**8**	**Steam Turbine Building Crane**

FIG. 10. Cross section of turbine building (Turbine Generator Set).

4. MAIN DATA OF THE OPTIMIZED EPR PLANT AND THE REFERENCES DESIGNS FROM WHICH THE EPR IS DERIVED

4.1. Data on EPR nuclear island

Reactor power	4900 MW_{th}
Steam generator power	4925 MW_{th}

4.2. Optimized EPR conventional island

– **For an EPR equipped with cooling tower:**

Electrical power, gross	**1850 MW**
Gross efficiency ($P_{gross}/P_{Reactor}$)	**37.8 %**
Net efficiency ($P_{net}/P_{Reactor}$)	**35.9 %**
Auxiliary power	approx. 5 %
Steam turbine type	1 double-flow high-pressure turbine section 3 double-flow low-pressure turbine sections
Low-pressure exhaust area	6 x 20 m²
Length of turbine generator set	approx. 60.5 m
Cooling system	natural-draft cooling tower
- temperature	18.2°C
Condenser pressure	0.060 bar
Generator apparent power	2056 MVA
- rotor cooling medium	hydrogen

– **For an EPR with direct cooling:**

Electrical power, gross	**1870 MW**
Gross efficiency ($P_{gross}/P_{Reactor}$)	**38.2 %**
Net efficiency ($P_{net}/P_{Reactor}$)	**36.3 %**
Auxiliary power	approx. 5 %
Low-pressure exhaust area	6 x 25 m²
Length of turbine generator set	approx. 63.5 m
Cooling system	direct cooling
-temperature	12 to 16°C
Condenser pressure	0.049 bar

4.3. Current situation

Due to recent discussions between German and French utilities the reactor power for designing the EPR is intended to be limited to 4500 MW, although the application for a license of the first EPR unit in France will be made with 4250 MW. The expected electrical power is some

– 1630 MW for a site equipped with cooling tower, and
– 1645 MW for a sea site with direct cooling.

The presented design optimization and economy advantages are still valid.

4.4. Reference designs from which EPR is derived

	German Konvoi Series	French N4 Series
Steam generator power	3867 MW_{th}	4270 MW_{th}
Electrical power, gross	1440 MW	1520 MW
Steam turbine type	HP-LP/single-shaft	HP-IP-LP/single-shaft
Low-pressure exhaust area	6 x 20 m^2 (Isar 2)	6 x 20 m^2
Generator	1640 MVA	1710 MVA
- rotor cooling medium	water	hydrogen
Length of turbine generator set	approx. 65 m	approx. 69 m
Feedwater pump drive	electric motor (~ 20 MW)	steam turbine
Gross efficiency ($P_{gross}/P_{Reactor}$)	**37.4 %**	**35.8 %**
Start of operation	in 1988 (Isar 2)	in 1998 (Chooz)

5. ECONOMY AND PROSPECT

The capital investment required to construct the European Pressurized Water Reactor could be kept below that of a Konvoi series plant. Even increased economy will be achieved due to the higher output of the EPR.

The reasons behind such remarkable cost reductions can be summarized as follows:

- The entire development process chain, from design all the way to subsequent plant maintenance work, has been thoroughly reviewed;
- Plant efficiency has been increased mainly through technical innovations to the turbine and optimization of the steam, condensate and feedwater cycle together with the heat sink;
- Plant systems and components have been optimized;
- Common general European codes are to be applied to the conventional island together with related national codes and standards;
- Design efforts have made optimum use of specialized personnel with global turnkey know-how i.e., specialists with know-how of both conventional and nuclear island design;
- There has been only moderate price escalation in Germany and around the world over the last 10 years due to streamlining, competition and globalization of the world market.

The result of all these optimized innovations and improvements is an advanced and competitive conventional island which, at minimum, fulfills all the technical objectives of the nuclear island spelled out under Section 1 above. It is well matched to the EPR nuclear island, but also to the nuclear islands of other nuclear power plant supplier designs, as site-specific adaptations such as power range, power grid structure, cooling system, twin units, etc., can be easily implemented.

The calculated power-generation cost of the overall plant is competitive with that of fossil-fueled power plants, and the EPR is much less dependent on fuel-cycle cost fluctuations due to the lower impact of nuclear fuel on the determination of power-generation costs.

Finally, it should yet be mentioned that nuclear power plants do not generate any carbon dioxide emissions. This constitutes an important contribution towards reaching the CO_2 emissions targets set at world climate conferences. Compared with a hard-coal-fired power plant, an EPR can save some 10,000,000 tons CO_2 emissions per year.

AP1000: Meeting economic goals in a competitive world

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Abstract. In the U.S., conditions are becoming more favorable for considering the nuclear option again for new baseload generation. While oil and natural gas prices have risen, the cost of operating the existing fleet of nuclear plants has decreased. Furthermore, an advanced 1000 MWe nuclear plant that will be even more cost-competitive with fossil fuels and natural gas will be available by 2005. Westinghouse, in an effort to further improve on the AP600's cost competitiveness, has developed the AP1000, a two-loop, 1000 MWe, advanced pressurized water reactor (PWR) with passive safety features and extensive plant simplifications to enhance the construction, operation, and maintenance. Like the AP600, the AP1000 uses proven technology that builds on over 30 years of operating PWR experience. Westinghouse has completed design studies that demonstrate that it is feasible to increase the power output of the AP600 to at least 1000 MWe, maintaining its current design configuration and licensing basis. To maximize the cost savings, the AP1000 has been designed within the space constraints of the AP600, while retaining the credibility of proven components and substantial safety margins. The affect on the plant's overnight cost of the increased major components that is required to uprate the AP600 to 1000 MWe is small. This overall cost addition is on the order of 11 percent, while the overall power increase is almost 80 percent. This paper describes the changes made to uprate the AP600 and gives an overview of the plant design.

1. THE CURRENT ENVIRONMENT FOR NUCLEAR ENERGY IN THE UNITED STATES

In recent years, the conditions that would enable the consideration of a new generation of nuclear energy plants have all been moving in favorable directions. These conditions include:

- Electric industry deregulation,
- Performance and safety of existing nuclear plants,
- More stable regulatory environment,
- Increased concern over greenhouse gases and global warming, and
- Low Cost Alternatives for New Baseload Generation in Light of Higher Fuel Prices.

1.1. Electric power industry deregulation

The operating nuclear plants have proven themselves to be valuable assets for reliably generating electricity at minimal costs, with minimal environmental impact, and without creating undue risks to the general public. While support for nuclear energy has waxed and waned over the past several decades, deregulation of the electric power industry is creating an environment that is proving very attractive for the existing nuclear plants – just the opposite of what many in the industry had anticipated.

The movement to deregulate electricity markets, initiated about a decade ago, is leading to major changes in the structure of the electric power industry, on a global scale. The basic premise of deregulation is that power generation should be unbundled from the transmission and distribution functions. Thus, generation can be left to an unregulated, competitive marketplace – while, transmission and distribution can continue to be the responsibility of regulated utilities.

Many industry experts feared that deregulation would lead to the demise of the current fleet of nuclear plants. To the contrary, however, it has, instead, produced a nuclear industry that is more viable than it has been in decades. Preparation for transitioning to a deregulated environment has forced all parties – including regulators, plant owners, and suppliers – to look closely at the economics of nuclear power generation, relative to alternative sources for electricity generation.

The forced sales of generating assets has been preceded by negotiations between the regulators and the utilities, concerning fair and equitable treatment of stranded investments. As a result, utilities have then been able to sell existing nuclear units at prices low enough to attract buyers. The units are being purchased by a relatively small number of unregulated power generation companies that are able to efficiently operate the units, because of the economy-of-scale associated with running a large number of reactors with one organization. Thus, the industry is now seeing the formation of large, efficient generating companies that could become potential customers for new nuclear plant capacity.

Besides consolidation of the plant owners, the nuclear industry is also seeing consolidation of the suppliers – reactor suppliers, architect/engineers, services companies, equipment vendors, and the like. The overall result of these consolidations is the formation of a healthy, viable industry that will be well positioned to efficiently satisfy the needs of a competitive marketplace.

1.2. Performance of the current fleet of nuclear power plants

Accompanying the industry's consolidation, the existing nuclear power plants have, in recent years, enjoyed substantial improvements in performance (for example, plant availability) and reduced operating costs (for example, reduced staffing levels). In many parts of the world, including the U.S., existing nuclear plants are able to generate electricity at lower costs than any alternative, except hydro. U.S. industry experts are saying that for the first time in more than a decade, production costs at U.S. nuclear power plants are the lowest of any major reliable electricity fuel source, even dropping below coal-fired plants. Production costs (which include fuel, operations and maintenance) at nuclear power plants averaged 1.83 cents per kWh in 1999, lower than coal's 2.07 cents and well below oil-fired plants at 3.18 cents and natural gas-fueled plants at 3.52 cents. And, this does not reflect last year's price spikes for oil and natural gas. Industry experts expect most, if not all, of the nation's 103 nuclear plants to extend their operating licenses for 20 years. But some utilities are taking a further look at nuclear power, particularly if they are able to build at existing sites and use a standardized design that could streamline the lengthy licensing process and cut construction expenditures.

1.3. Safety record

The existing nuclear plants around the world have amassed an impressive safety record that provides another reason for growing public acceptance. With the exception of the Chernobyl accident (which was based upon a technology no longer offered in the nuclear industry), the health and safety of the general public has never been threatened by nuclear energy. After more than a quarter century of safe operation, coupled with their clean air and economic benefits, the operating nuclear units are gaining greater and greater acceptance by the general public.

1.4. Regulatory stability

A major factor in holding down operating costs has been the stabilization of the safety regulatory environment. After more than a decade of changes mandated by the safety regulators, because of accidents such as the one at Three Mile Island, the regulatory environment has improved significantly. Recognizing that regulatory stability is a critical component in assuring economic competitiveness of the existing nuclear plants, the U.S. Nuclear Regulatory Commission is transitioning to a risk-informed, performance-based regulatory structure that is proving to be highly effective and efficient. As a consequence of such changes, the investment community is becoming more optimistic about nuclear energy's viability. On top of this, the plant owners are extending the lifetimes of their operating units and obtaining regulatory approval for doing so. As a result, there is a growing sense of optimism that the existing nuclear industry will be viable for decades to come.

1.5. Environmental benefits

Meanwhile, there is a growing concern about the effects that greenhouse gases will have on the earth's environment. The international community is moving toward a consensus (through agreements such as the Kyoto treaty) that significant steps must be taken to reduce the production of greenhouse gases. In fact, the U.S. Energy Information Administration has noted that the largest single contribution to avoiding new greenhouse gas production in the U.S., in recent years, has been the increased availability and power uprating of the operating nuclear units. Many environmentalists and governments are beginning to acknowledge that the existing nuclear plants are playing an important role in avoiding further production of greenhouse gases and other air pollutants. As a result, there is growing support for extending the lifetimes of the existing nuclear units.

1.6. Low cost alternatives for new baseload generation

Market analysis of the U.S. electricity generating market indicates that the generating cost of competitive new generating capacity must be less than $0.03 per kWh. When such factors as an attractive return on investment and payback period are considered for a new nuclear electric generating facility, this results in the requirement to have an overnight capital cost of approximately $1000/kW or less. Industry executives indicate that any new nuclear plant must be able to compete in the deregulated generation wholesale marketplace and provide a return to the shareholders. Against this standard, the costs of advanced nuclear power plants currently available are still too high. This includes the AP600, Westinghouse's 600 MWe advanced, passive plant, which was issued Design Certification by the U.S. NRC at the end of 1999.

When the AP600 was developed, the U.S. Utility Requirements Document for advanced light water reactor plants included a cost goal that was based on the cost of coal and natural gas generated electricity at the time the document was written. The AP600 meets this cost goal; the overnight capital cost for the first AP600 plant is calculated to be between 1300-1500 $/kW depending on the site selected. Although the AP600 is the most cost-effective nuclear power plant ready for deployment, it is still more expensive than other new generation options in the U.S.

In response to the demand for a clearly cost-competitive nuclear power plant, Westinghouse developed the AP1000. Westinghouse has completed design studies that demonstrate the feasibility of increasing the power output of the AP600 to at least 1000 MWe, while maintaining its current design configuration, use of proven components and licensing basis.

2. AP1000 – COST COMPETITIVE NUCLEAR OPTION

The AP1000 builds on the design and licensing basis of the AP600, while providing higher power output without an appreciable increase in capital cost. This is achieved by designing 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 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 and 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. Figure 1 shows a section view of the AP1000 and AP600 containments; Figure 2 shows a plan view.

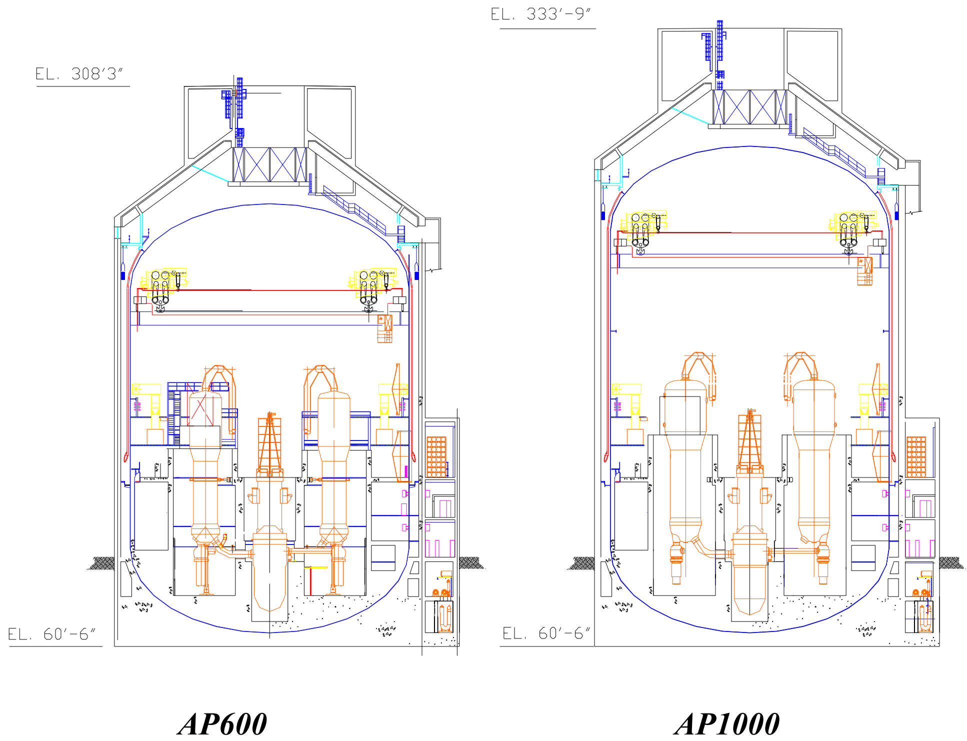
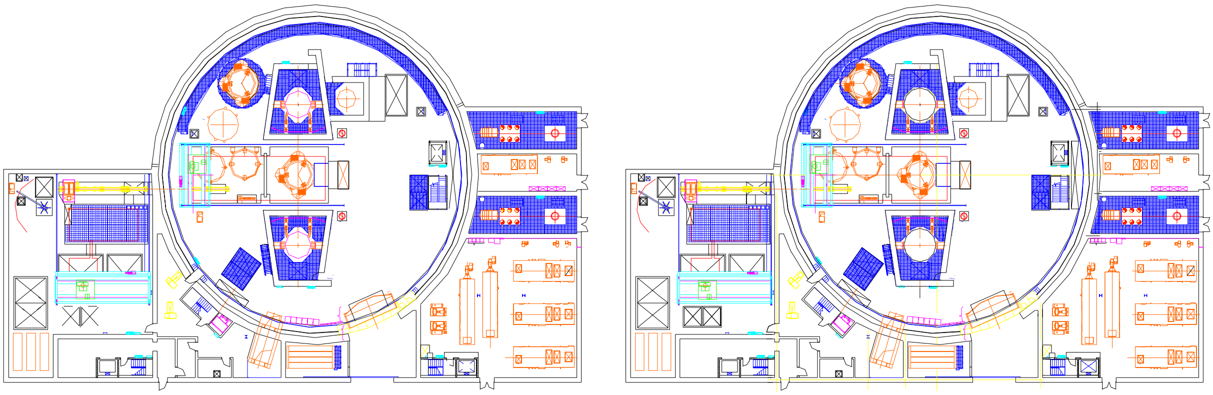


FIG. 1. Westinghouse AP1000 and AP600 plants (section).



AP600 ***AP1000***

FIG. 2. Westinghouse AP1000 and AP600 plants (plan).

The affect of the changes that are required to uprate the AP600 to the AP1000 is small on the plant's overnight cost. A detailed estimate of each difference from AP600 was applied to the already extensive and validated AP600 cost estimate. This overall cost addition is on the order of 11 percent. The overall power increase, however, is almost 80 percent. This yields a greatly reduced overnight cost per megawatt.

2.1. Larger components

To achieve the higher power level, the following changes have been incorporated into the basic design of the advanced passive plant:

- Increased core length and number of assemblies
- Increased size of key NSSS components
  - Taller reactor vessel
  - Larger steam generators
  - Larger canned reactor coolant pumps
  - Larger Pressurizer
- Increased containment height
- Some capacity increases in passive safety system components
- Turbine Island sized to increase power rating

Like the AP600, the AP1000 plant is designed to be simple to construct, operate and maintain with significantly fewer safety and non-safety components, simpler components, and better materials than a currently operating PWR.

The design of the major components used for power generation (fuel, internals, steam generator, reactor coolant pumps, turbine, etc) is based on equipment that has successfully operated in power plants. Modifications to these proven designs are based on similar equipment that has had successful operating experience in similar or more severe conditions.
Table I provides a summary comparison of the key design parameters of the AP1000 with those of the AP600.

2.2. Core design

The major differences in the AP1000 core design compared to the AP600 core design are the addition of 12 fuel assemblies, an increase in the length of the fuel assemblies, and additional control assemblies. The extra assemblies and increase in length along with an increase in the linear power density in the core enables the core power rating to be increased from 1,933 MWt to 3,400 MWt within the same diameter reactor vessel. The number of rod control cluster was increased to 53 in the AP1000 compared to 45 in the AP600. The AP1000 core also incorporates the Westinghouse ROBUST fuel assembly design compared to the Vantage 5-H design of the AP600. The ROBUST design includes guide tubes with increased wall thickness.

TABLE I. COMPARISON OF NSSS DESIGN PARAMETERS

	AP600	AP1000
Reactor Power, MWt	1933	3400
Net Electric Output, MWe	610	1090
Hot Leg Temperature, °F	600	615
Number of Fuel Assemblies	145	157
Type of Fuel Assembly	17x17	17x17
Active Fuel Length, ft	12	14
Linear Heat Rating, kW/ft	4.10	5.707
R/V I.D., inches	157	157
Number Control Rod Assemblies	45	53
Hot Leg / Cold Leg Pipe ID, in	31/22	31/22
Steam Generator Heat Transfer Area, ft^2	75,180	125,000
Reactor Coolant Pump Flow, gpm	51,000	75,000
Pressurizer Volume, ft^3	1600	2100
Core Makeup Tank, # / Volume, ft^3	2/2000	2/2500
Containment Diameter / Height, ft	130/190	130/215

2.3. Reactor vessel

The AP1000 reactor vessel has the same overall diameter and number and size of nozzles as the AP600 vessel. The overall length of the AP1000 vessel has been increased to accommodate the increase in core length to 14 feet. The AP1000 reactor vessel internals are essentially the same design as the AP600 vessel internals, with the major differences being that the length of the lower internals has increased because of the longer core design. Also, the thickness of the lower support plate has increased to accommodate the heavier AP1000 core, which has both additional fuel assemblies (12) and heavier assemblies due to the longer length. The AP1000 integrated head package design is the same as that of the AP600 except that the overall height has increased to accommodate the longer control rod drives and incore components required for the 14-foot AP1000 core. Internally, the AP1000 integrated head package also accommodates eight additional control rod assemblies. Figure 3 illustrates the overall height differences between the AP1000 and AP600 reactor vessels and integrated head packages.

2.4. Steam generators

The AP1000 Model Δ125 steam generators incorporate very similar features to the AP600 Model Δ75 SGs. Both units are vertical-shell U-tube evaporators with a triangular pitch tube bundle and integral moisture separating equipment. They both use Inconel-690 thermally treated tube material. To accommodate the higher thermal output of the AP1000 more heat transfer surface is required, thus increasing the shell diameter and height to enclose the larger tube bundle and larger moisture separation equipment required for the higher steam flow. The mass of water stored in the secondary side AP1000 steam generator has been increased such that it is about 36-percent larger, on a per megawatt basis, than that of the AP600 steam generator. This increased water mass results in a greater heat transfer capability from the reactor coolant system during transients and improves safety margins. Westinghouse has successful experience in building and operating steam generators as large as the Δ125s in a number of plants including Arkansas, San Onofre and Waterford. Figure 4 illustrates the dimensional differences between the AP1000 and the AP600 steam generators.

2.5. Canned-motor reactor coolant pumps

The same basic canned-motor pump design is employed in the AP1000 as in the AP600 including the use of a uranium alloy flywheel to provide rotating inertia to extend the flow coastdown. However, the higher thermal power and core power density of the AP1000 requires higher flow and longer coastdown from the AP1000 pumps compared to the AP600 pumps.

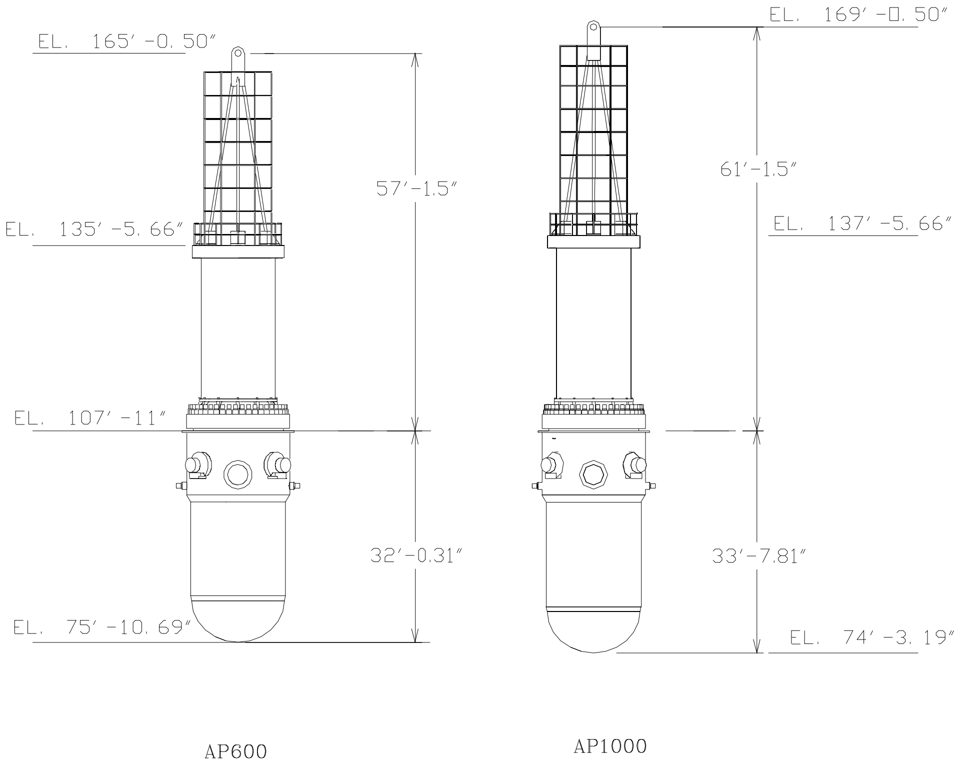


FIG. 3. AP1000 and AP600 reactor vessel and upper head package.

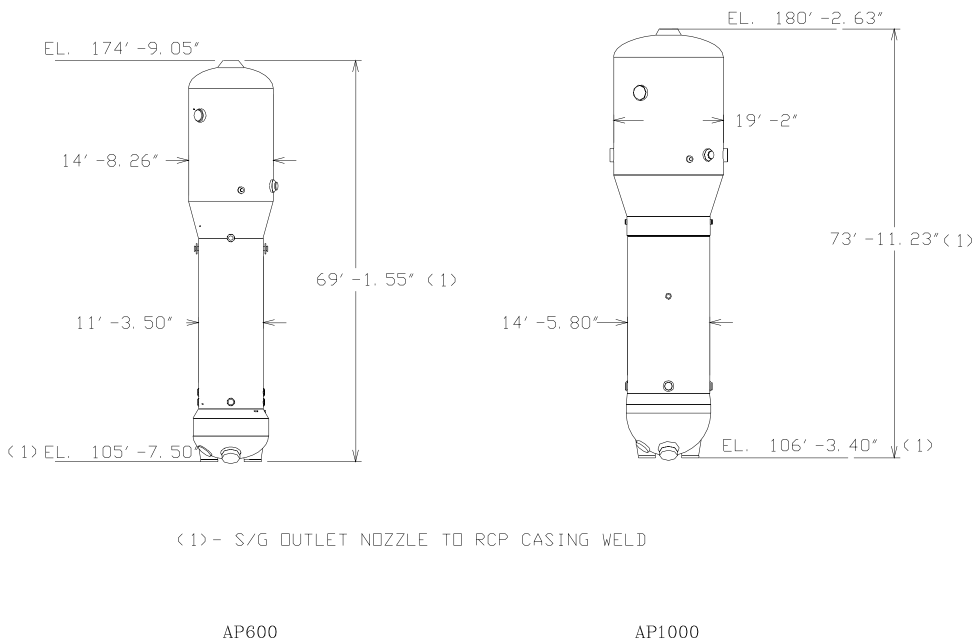


FIG. 4. AP1000 and AP600 steam generators.

2.6. Pressurizer

The AP1000 pressurizer volume was increased compared to the AP600 to accommodate the larger reactor coolant system volume in the AP1000. This was accomplished by making the AP1000 pressurizer taller while maintaining the same diameter pressurizer as in the AP600. The total volume of the AP1000 pressurizer is 2,100 ft^3 compared to 1,600 ft^3 for the AP600.

2.7. AP1000 passive safety features

The AP1000 passive safety features use the same design approach and arrangement as the AP600 (Figure 5). The capacities of the AP1000 passive safety features have been selectively increased using insights from the AP600 design, testing, analysis and licensing activities. Two key factors in these insights are the uncertainty in the computer analysis tools and the margin between the calculated results and the licensing limits. These insights indicate that whereas some passive safety features should be increased at least as much as the increase in core power, other features do not need to be increased as much.

2.8. AP1000 accident analysis

Preliminary accident analyses have been performed for the AP1000 using the AP600 validated analysis codes and preliminary models of the AP1000 plant. This report has been prepared and submitted to the U.S. NRC as part of the pre-application review of the AP1000. These analyses are not a complete set of analyses as prescribed by 10 CFR 50. They were performed to characterize the expected performance of the AP1000. These analyses were performed using bounding assumptions and are performed in a manner consistent with the approach taken for the AP600.

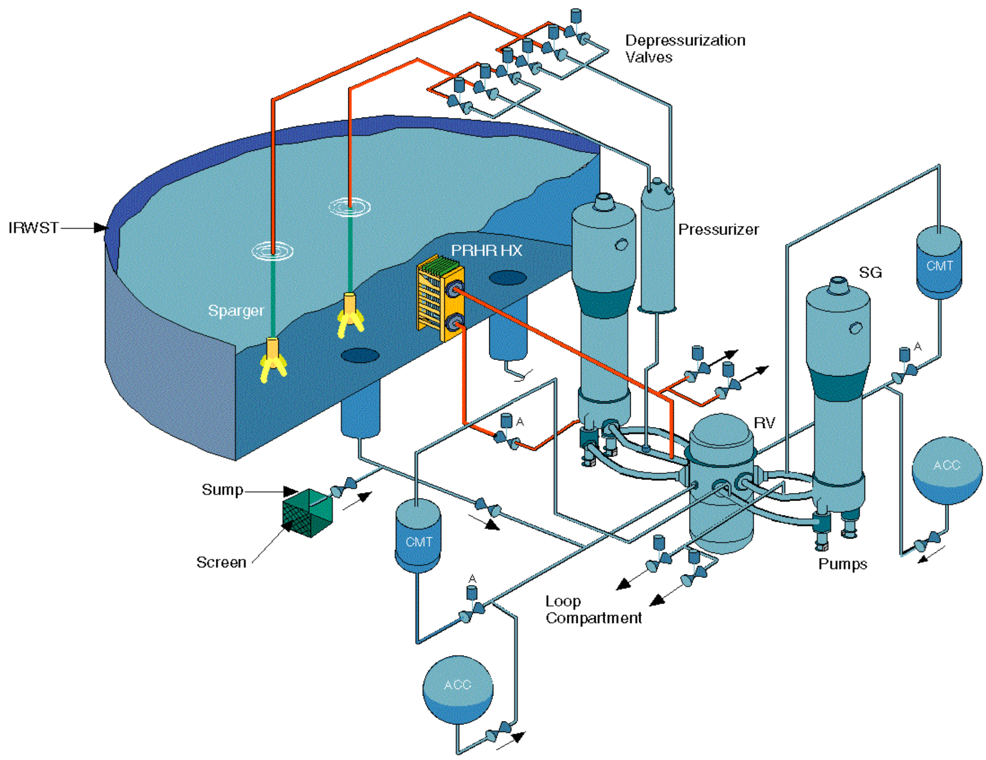


FIG. 5. AP1000 reactor coolant system and passive core cooling system.

Based on the results of these analysis assessments, it appears that the analysis results for the AP1000 will provide large safety margins for the range of postulated accidents and transient events. The timing and interactions predicted for the AP1000 are similar to the performance predicted for the AP600. No new phenomenon or significant differences in performance characteristics were observed in the analysis results.

Westinghouse has evaluated scaling studies of the AP600 tests and assessed their scalability to the AP1000. This report has been prepared for submittal to the U.S. NRC as part of the pre-application review of the AP1000. This evaluation indicates the AP1000 design features and operating characteristics have been selected such that the performance of each safety feature and their interdependent effects are judged to be sufficiently similar to the AP600 such that the AP600 test data should satisfy the NRC requirements in 10 CFR 52.47. As a result, the Westinghouse analysis codes validated for AP600 should be sufficient to perform the accident analyses for Design Certification of the AP1000 without the need for additional testing.

2.9. Costs and construction

The change in size of the affected components does not impact the construction schedule for the AP1000. And, there are less than ten additional valves required. The increased containment height is accommodated in the existing containment module rings, so no additional containment ring lifts are required. The only addition to the construction schedule is one more shield building concrete pour. It is on the critical path and will add at most two weeks.

There is no change to the plant's availability potential or staffing level since they are independent of power level.

Table II compares the cost to build, operate and decommission the AP600 and AP1000 plants. Evaluation of the generation costs for the AP1000 yields results of approximately 3 ¢/kWh for twin units constructed on a single site. This makes AP1000, in general, more than competitive with generation using fossil and renewable fuels.

TABLE II. AP600 VS. AP1000 ECONOMICS IN THE U.S.*

Aspect	AP600	AP1000
Cost to Build ($/kWe)	1400	900-1000
Cost to Build (cents/kW-hr)	2.9	1.9
Cost to Operate (cents/kW-hr)	1.3	1.0
Cost to Decommission (cents/kW-hr)	0.1	0.1
Total Generating Cost (cents/kW-hr)	4.3	3.0

* For 20-year financing at commercial rate of return.

2.10. AP1000 licensing status

The AP1000 is being designed to meet U.S. NRC regulatory criteria in a similar manner to that found to be acceptable for the AP600. Furthermore, the AP1000 is being designed to meet NRC deterministic safety criteria and probabilistic risk criteria with large margins. Westinghouse intends to certify the AP1000 standard plant design under the provisions of NRC regulatory criteria 10 CFR Part 52.

Preliminary pre-application discussions with the NRC began in 2000. Westinghouse continues in discussions with the NRC to identify areas of review that are necessary to obtain Design Certification for the AP1000 beyond those already accepted by the NRC for the AP600 Design Certification. Interest in AP1000 has led to support from the U.S. Department of Energy and the Electric Power Research Institute, primarily to facilitate completion of the plant's safety case review.

The strategy for licensing the AP1000 is to leverage the AP600 licensing program. To that end, Westinghouse is working to obtain NRC agreement on a plan to avoid high cost activities during Design Certification review, such as additional testing, safety code development, and additional detailed engineering. To further support this strategy, Westinghouse has evaluated scaling studies of the AP600 tests, assessed their scalability to the AP1000, and has submitted the evaluation to the NRC. The current schedule proposed under consideration is for obtaining Design Certification of the AP1000 by the end of 2004.

2.11. Looking beyond AP1000 design certification

As noted above, the licensing strategy for AP1000 is to use the original licensing bases that were applied to AP600, so as to minimize the cost and schedule for obtaining NRC certification and, thus, bringing AP1000 to the marketplace as quickly as possible. The U.S. Department of Energy is funding some of the analytical work on the AP1000 design.

Meanwhile, however, Westinghouse is currently involved in three other research & development projects being funded under the U.S. Department of Energy's Nuclear Energy Research Initiative. These projects include: (1) risk-informed assessment of regulatory and design requirements for future nuclear plants, (2) development of methodologies for utilizing smart equipment with self-monitoring, self-diagnostic features, and (3) development of advanced technologies for design, fabrication, and construction of future nuclear plants. If future funding becomes available, it should be possible to apply the results of these efforts to the AP1000 design, as a means to further reduce costs and improve performance. However, any such changes would not be pursued until after the AP1000 design has been certified by NRC. Bringing the design to the marketplace as quickly as possible must be the highest priority. Further improvements can follow.

3. CONCLUSIONS

The AP1000 is derived directly from the AP600, which uses passive safety features and extensive simplifications to enhance construction, operation, and maintenance. Design changes related to uprating the AP600 to 1000 MWe are being incorporated into the AP1000 standard plant design that Westinghouse intends to license in the U.S. under 10 CFR Part 52. The AP600 design has already been licensed with the NRC, receiving Design Certification in December 1999.

Preliminary safety evaluations and analysis results, performed on the AP1000, indicate that passive safety features can be successfully applied to a plant of a higher power rating while maintaining large safety margins. Scaling evaluations indicate that the AP600 test program and the analysis codes validated for AP600 should be sufficient to perform the accident analyses for Design Certification of the AP1000 without the need to perform additional testing.

The design evaluations performed on the AP1000 indicate that the design objectives of maintaining the AP600 design configuration, use of proven components and licensing basis can be met and that the AP1000 costs will be competitive in the U.S as well as other parts of the world.

Optimization of design solutions on safety and economy for power unit of NPP with WWER reactor of new generation

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Abstract. Development of new generation WWER reactors is being carried out in Russia. These new projects with WWER reactors aim to achieve increased levels of safety and reduced costs. This paper describes these designs and discusses the main factors leading to the safety level increase and the improved economics.

1. INTRODUCTION

Since 1988 the projects of NPP power unit with WWER reactors of new generation have been developed. Their construction and commissioning is expected within the period after 2000. Nowadays the project of power unit with WWER-1000/V-392 reactor has been developed and licensed in Gosatomnadzor of RF.

The construction of two power units under this project is expected to start at Novoronezhsky site and basic solutions on safety are used in the project "Kudankulam" in India and in a 1500 MW nuclear power plant (WWER-1500) being developed nowadays.

The concept of new projects of NPP with WWER reactors is aimed at the achievement of two main targets:

- Increase of safety level;
- Increase of power production efficiency and cuts of expenses for construction and commissioning.

Below is presented a brief description of the basic design solutions resulting in realization of these targets.

2. DESIGN SOLUTIONS ON SAFETY LEVEL INCREASE

Design solutions on safety for power unit of NPP with WWER reactor of new generation are aimed at erection of NPP with higher safety level to minimize the risk of NPP use to a reasonable possible level. In doing so all requirements of current normative documents on safety, accepted in Russia, as well as recommendations of IAEA. In particular an assignment of the requirements for design values of probabilistic safety parameters in NV NPP –2 project is based on the requirements of Item 1.2.17 of OPB-88/97 in accordance with which a value of limiting accidental release frequency should not exceed a value of 1.0E-7 / per reactor year. Maximum accidental release (MAR) is a release of such amount of radioactive products which will demand the population evacuation beyond the distances specified by current norms of NPP disposition /2/.

The second requirement for risk level limitation is a requirement of Item 4.2.2 of OPB-88/97 in accordance with which a core damage (CD) frequency should not exceed 1.0 E-5 / per reactor year.

It should be noted that the frequencies of MAR and CD are evaluated according to the results of probabilistic safety assessment (PSA) of the first and second level as cumulative frequency values for all accident sequences which can result in such occurrences during internal (equipment and components failure), on-site (fire, flooding, etc.) and external events and impacts (of natural and technological character).

Development of design solutions on safety increase in the project of NV NPP-2 is based on realization of basic principles of defense-in-depth concept which includes the erection of several physical barriers to prevent radioactive release and high level reliability of these physical barriers to protect them from damage. When developing design solutions for NV NPP-2 the experience in design and operation of a serial unit of NPP with WWER-1000/V-320 has been considered. Nowadays in Russia, Ukrane and Bolgaria 14 power units with V-320 reactors are in operation and power units in NPP "Temelin" (Chechia) and in Rostov (Russia) will be commissioning in 2000. Use of NPP with V-320 reactors offered the prospect of achievement of high level safety as during the total period of these power unit operation (270 reactor/year) not a single serious accident has occurred. But the shortages, revealed through results of NPP experience and results of probabilistic safety assessment, showed the necessity of improvement and modifications to be implemented into reactor facility in order to meet all requirements of current defense-in-depth concept including decrease of core damage frequency and maximum emergency release frequency.

A new reactor facility has been used in NV NPP-2 project and new safety structure has been developed.

2.1. Reactor facility

In comparison with the V-320 reactor the following improvements have been made in the project NV NPP-2:

- The efficiency of mechanical system of reactor emergency protection has been increased providing fast transition of the reactor into a subcritical state and maintenance of this state up to temperature less than 100-120 °C without boric acid supply. This has been achieved by increase of the number of operating components from 61 for V-320 to 121 for V-392;
- The system of automatic suppression of xenon oscillations has been developed;
- A new main cooling water pump ГЦН-1391 has been used in which water is used for lubricating and cooling of bearings and in which the strength seals is increased and which can operate without damage during not less than 24 hours under conditions of cooling loss;
- Steam generator design has been improved providing considerable leakage frequency decrease through heat exchangers and collectors of steam generators;
- Core design has been improved allowing to increase the reliability level and reduce its component damage;
- Safety valves are used capable to operate with use of steam and water mix;

- Specific measures are implemented to protect from damage the boundaries of reactor coolant system and associated components including use of constructional materials, observation of operation requirements and control of reactor body state, equipment and pipelines during operation as well as necessary strength margins. Reliability of the reactor coolant system boundaries is justified by experience and results of specific design strength analysis including the evaluation of leakage occurrence and equipment and pipeline damage frequencies on the basis of probabilistic and strength models. In particular the reactor design shows that the calculated frequency value of a reactor body break does not exceed 1.0 E-7 1/year within the increased period of operation since 30 years for V-320 to 60 years for V-392.

2.2. Safety system

Protection of physical safety barriers which are fuel for NPP with WWER (fuel matrix and fissile elements), the boundary of reactor coolant circuit and containment is provided by use of engineering systems intended for execution of the following safety functions:

- Reactivity control that is the reactor transition into a subcritical state and maintenance of this state for all operating parameter range;
- Heat removal from fuel in a core and spent fuel in a fuelling pool;
- Maintain of coolant store in a core during LOCA accidents;
- Limitation of radioactive release into environment.

Achievement of high level reliability of safety functions in NV NPP-2 project is based on use of basic engineering principles and requirements for structure and construction of safety systems, presented in OPB-88/97 and INSAG-3 which are supported by results of qualitative analysis of reliability and PSA of 1 and 2 levels. Special attention has been paid for the following basic principles during development of safety structure systems:

- Protection from common cause failure (CCF);
- Extended use of passive systems;
- Use of functional and structural variety;
- Protection from human errors;
- Protection from internal and external impacts.

It should be noted that, as PSA showed, not sufficient development of these principles in the project of NPP with WWER-320 is the main reason of level safety limitation for existing unit of NPP with WWER-1000.

As a result of realisation of these principles in NV NPP-2 project the safety system structure has been developed based on use of mutual reserving and fully independent active and passive safety systems each is able to execute each separate safety function in full measure. The detailed list of safety functions is given in Table I. with the list of active and passive systems intended for execution of each safety function. The principal structural schemes of safety systems in NV NPP-2 project are given in Fig.2.2-1 and 2.2.-2.

Use of mutual reserving active and passive systems allows to provide the high level reliability of function execution due to reduction of common cause failure (use of functional and structural variety) and due to reduction of human errors (operation of passive systems does not require operator's actions).

Additionally auxiliary measures to reduce impact of common cause failures and human errors are used in the active system design.

TABLE I. FUNCTIONS AND SAFETY SYSTEMS

Safety functions	**Safety systems**	
	Active	**Passive**
1.Transfer the reactor into a subcritical state and keep this state in a range of operating parameters	Reactor emergency protection system with 121 operating mechanisms	Fast boron injection system
2. Heat removal from reactor through the 2-nd circuit	Four-channel system of emergency heat removal through a steam generator with the structure of 4 x100% (one channel is able to execute functions in full measure). 2 channels of the system are used during normal operation for coolant purification of the secondary circuit. Two channels are in standby mode.	Four-channel passive heat removal system through steam generators with the structure 4x33% (three channels are able to execute functions in full measure).
3. Maintenance of coolant store in a core during LOCA	Four-channel system of emergency core cooling with the structure of 4 x100%. 2 channels of the system are used during normal operation for heat removal from spent fuel in fuelling pool. Two channels are in standby mode. The system operates in pressure range of 0.1-8.0 Mpa in the primary circuit	Hydraulic tanks of the first stage with the structure 4x33% and pressure of 6.0 Mpa and water capacity 50 m^3 in each tank. Hydraulic tanks of the second stage with the structure 4x33% and water capacity designed for maintenance of core coolant store during 24 hours in the situation when the active system failed completely.
4. Isolation of steam generators from main steam collector	Each steam generator has fast-acting isolating valves with electric drives.	
5. Limitation of pressure in the primary circuit.	Safety valves in the pressurizer capable to operate both as active and passive systems.	
6. Limitation of pressure in steam generators and the secondary circuit.	Fast-acting reduction systems for steam release to air.	Safety valves of steam generators
7. Confinement of radioactive products inside the containment	Four-channel sprinkling system. The system of isolating valves of the containment. Ventilation system and clean-up system in annulus space between internal and external containment	Double containment of full pressure. Passive system of hydrogen removal. The trap system for melt fuel.

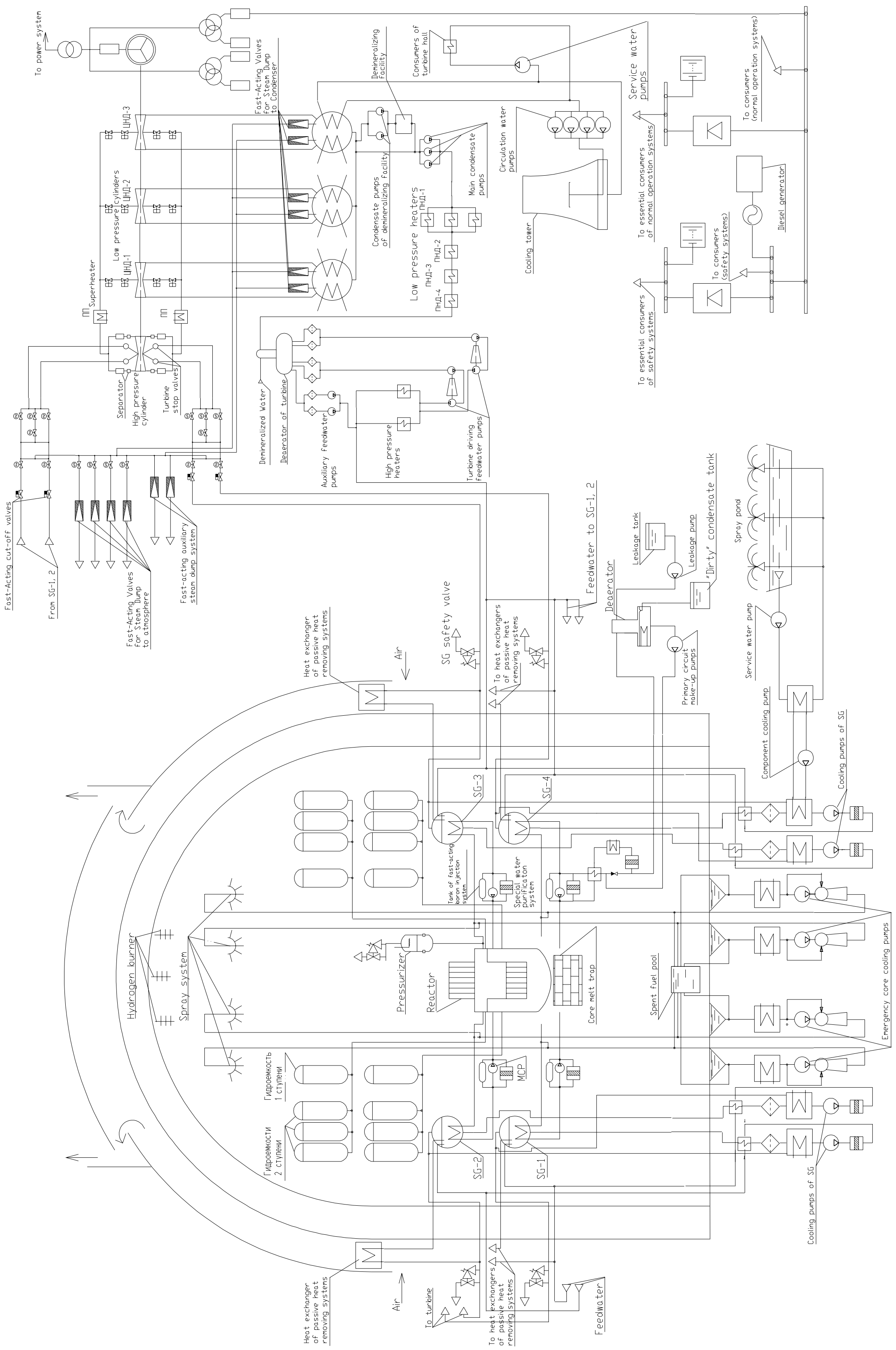
To power system
Fast-Acting Valves For Steam Dump to Condenser
Demineralizing facility
Consumers of turbine hall
Service water pumps
To consumers (normal operation systems)
To essential consumers of normal operation systems
Diesel generator
To consumers (safety systems)
To essential consumers of safety systems
Cooling tower
Circulation water pumps
Main condensate pumps
Condensate pumps of demineralizing facility
Low pressure heaters
ПНД-1
ПНД-2
ПНД-3
ПНД-4
Low pressure cylinders
ЦНД-1
ЦНД-2
ЦНД-3
Superheater
ПП
Separator
High pressure cylinder
Turbine stop valves
Demineralized Water
Deaerator of turbine
Auxiliary Feedwater pumps
High pressure heaters
Turbine driving feedwater pumps
Fast-Acting cut-off valves
From SG-1, 2
Fast-Acting Valves For Steam Dump to atmosphere
Fast-acting auxiliary steam dump system
Feedwater to SG-1, 2
Leakage tank
Leakage pump
Deaerator
"Dirty" condensate tank
Spray pond
Service water pump
Primary circuit make-up pumps
Component cooling pump
Cooling pumps of SG
Heat exchanger of passive heat removing systems
Air
SG safety valve
To heat exchangers of passive heat removing systems
SG-3
SG-4
Tank of fast-acting boron injection system
Special water purification system
Hydrogen burner
Spray system
Pressurizer
Reactor
Core melt trap
Spent fuel pool
Emergency core cooling pumps
Гидроемкости 1 ступени
Гидроемкости 2 ступени
MCP
SG-2
SG-1
Cooling pumps of SG
Heat exchanger of passive heat removing systems
Air
To turbine
To heat exchangers of passive heat removing systems
Feedwater

Fig. 2.2-1.

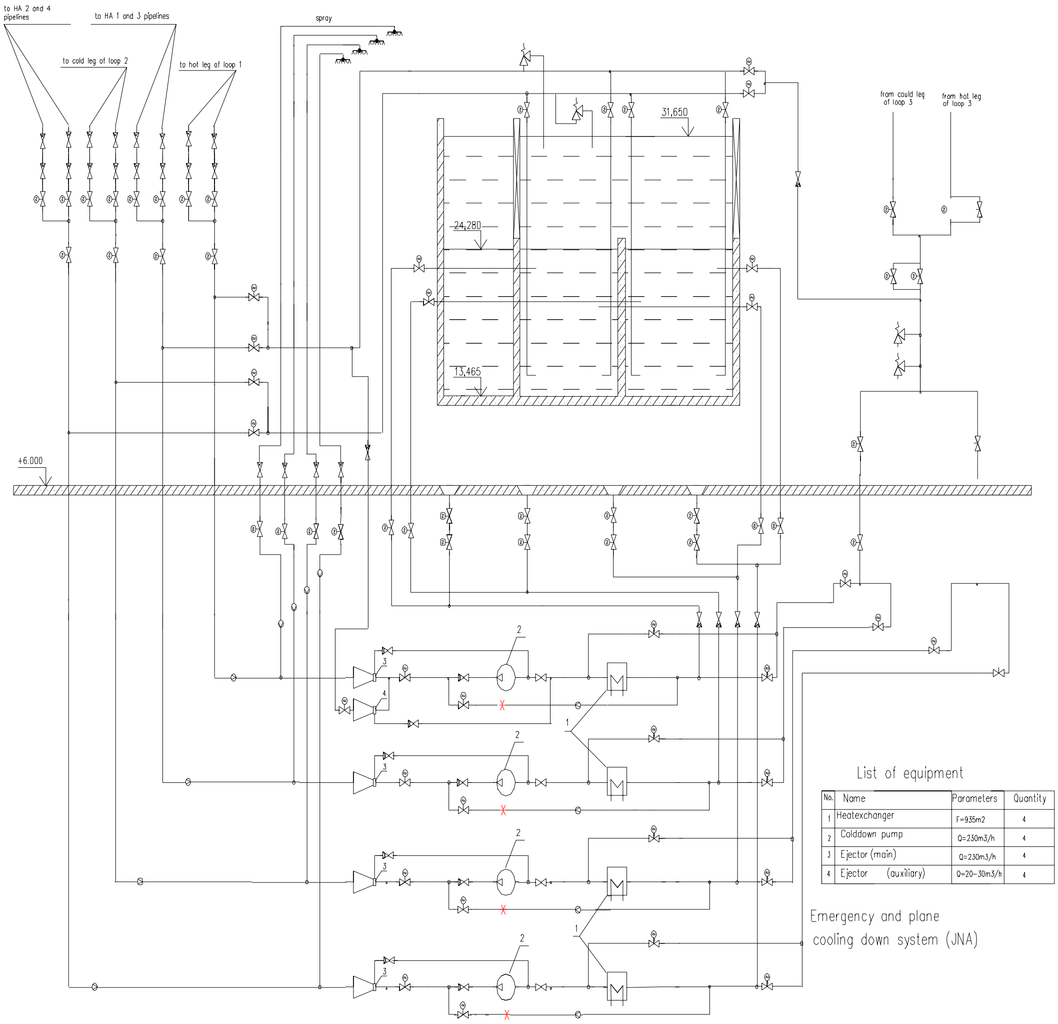

Fig. 2.2-2.

As an additional measure for protection from common cause failure the use of separate channels of emergency core cooling and heat removal systems through the secondary circuit for normal operation is provided. Two out of four channels of these systems operate permanently and two others are in standby mode during unit power operation. Different modes of operation and different state of the equipment provide additional protection from common cause failures.

It should be noted that most part of the equipment of operating channels (pumps, valves etc.) are in the same state which is required for execution of the specific functions during emergency situations. Such solution allows to increase the level of safety systems readiness due to exclusion of concealed failures of operating components and provide additional protection from common cause failures due to various modes of component use.

Protection from human errors for active safety systems is provided due to more high level of automated control when their operation is needed during transition and emergency situations as well as due to passive systems use not requiring control actions.

Use of hydro accumulators of the 2-nd level capable to sustain coolant store in a core under LOCA during 24 hours gives the possibility to extend time for control of accidents beyond the design basis related to LOCA accidents resulted in complete failure of active system of emergency cooling.

Use of double reinforced concrete containment with a passive system of hydrogen removal, ventilation system and air purification in annulus gap space between the primary and secondary containment, sprinkling system and melt core retain system (a trap for melt fuel) reduces release and size of sanitary and protective area for design accidents and prevents exceeding of emergency release for accidents beyond the design basis including large-scaled accidents involving complete melt of fuel.

2.3. Results of PSA

The comparison of contribution in frequencies of MAR from different groups of internal initiating events (IE) for NV NPP-2 unit No. 1 with a reactor V-392 and unit 4 of Balakovsky NPP with the reactor V-320 is given in Table II. It should be noted that the results of frequencies of MAR for two units, given in the table2.3.-1, were obtained mostly with the use of the same initial data on reliability of components, probability of human errors and IE frequencies. Thus, the comparative analysis of the results is quite correct. Mainly it gives principal differences in design solutions in structure, principles and modes of safety system operation between the projects of NV NPP-2 and NPP with V-320.

TABLE II. CORE DAMAGE FREQUENCY FOR INTERNAL INITIATING EVENTS

Initiating	IE	CDF for NV NPP		CDF for Balakovo NPP	
Event (IE)	Frequency	Absolute 1/year	Relative %	Absolute 1/year	Relative %
SLOCA	3,20E-03	1,26E-09	~2,6	3,40E-07	~0,8
MLOCA	1,00E-03	3,64E-10	<1	8,30E-08	~0,2
LLOCA	3,20E-04	6,79E-10	~1,4	5,40E-08	~0,1
Leakage from primary to secondary curcuit	1,00E-03	1,26E-09	~2,6	1,10E-06	~2,6
General transients	1,00E-00	7,38E-09	~15	1,65E-06	~3,9
Loss of normal heat removal	1,00E-01	7,38E-09	~15	6,50E-07	~1,5
Loss of offsite power	1,00E-01	7,91E-09	~16	3,54E-05	~82,9
Nonisolable steam line leakage	1,00E-03	2,67E-11	<1	3,40E-06	~8,0
Isolable steam line leakage	4,00E-04	1,29E-10	<1	1,00E-10	~0
Loss of heat removal at shutdown	3,50E-05	1,07E-08	~22		
Loss of offsite power at shutdown	3,70E-03	1,12E-08	~23		
All IE		4,77E-08	100	4,27E-05	100

It is also should be noted that the solutions used in the project NV NPP-2 were based on the results of PSA for NPP with V-320. Table II. shows that cumulative frequencies of PSA for 7 groups of IE during reactor power operation are within 2.58xE-8 1/year for NV NPP-2 and 4.26x10-5 for unit 4 of Balakovsky NPP, that is the frequencies of PSA for the unit 1 of NV NPP-2 are approximately 1700 times less than for the unit 4 of Balakovsky NPP. Contribution into frequency of PSA from standby modes is approximately 2.2 E-8 1/ year for NV NPP-2.

Main contribution in MAR frequency for the unit 4 of Bal. NPP is made by initial events with power loss (83%), steam generator leakage in a part cut off from steam generator (8%), reactor outage (3.8%) and violation of heat removal in the secondary circuit (1.5%). Total contribution in MAR frequency is leakage from the primary into the secondary circuit (2.6%). Contribution of the primary circuit leakage inside the containment is 1%.

The main reasons of dominating contribution in MAR frequency because of initial events without the primary circuit leakage for NPP V-320 are as follows:

- Comparatively high frequency of realisation of such IE in comparison with primary circuit leakage;
- Comparatively low level of reliability of heat removal system from the secondary circuit and emergency power supply system from diesel-generators. It is explained by the fact that active three-channels safety systems are used in the project of NPP with B-320 based on the use of the components of the same design (diesel-generators, pumps, valves, return valves etc.) in separate channels of the safety system. Main contribution into indices of not readiness is made by common cause failures of the components of the same design and failures of diesel-generators;
- Comparatively low level of protection from human errors. For execution of main safety functions of long heat removal from the primary and secondary circuit as well as to control accidents beyond design basis (for instance, the mode of bleed and feed) the operator' actions are needed.

The main contribution in frequency decrease for unit No1 of NV NPP-2 in comparison with NPP V-320 is reached because of use of the following principally new design solutions:

- Use of mutual reserving passive and active systems for execution of main safety functions;
- Modified reactor emergency protection system with double number increase of operating mechanims in comparison with V-320 and fast-acting boron injection system for reactor transition \ into a subcritical state and keep this state in a wide range of operating parameters (maintenance of a subcritical state up to the temperature of 100° C);
- Active and passive systems of emergency heat removal in the secondary circuit. Both systems are able to remove heat during unlimited period of time while for NPP with V-320 CAP can operate during a limited period of time (about 30-40 hours) depending on the coolant stored in the tanks;
- Active system of the core emergency cooling system (ECCS) and hydraulic tanks of the first and second stages to maintain coolant in a core during the primary leakage. The hydraulic tanks of the second stage together with hydraulic tanks of the first stage reserve ECCS to maintain a core coolant during 24 hours after accident. This period of time can be used to renew serviceability of active ECCS during its failure.

It should be noted that the use of functional and constructional differences in safety system design allows to create the reliable protection from common cause failure and use of passive and active systems not requiring intervention of any operator allows to create in-depth protection from human errors.
2) Use of separate channels of active safety systems (CAP and ECCS) for normal operation. In doing so the most part of the components of these systems are in the same state as the states needed for execution of the required functions. Use of such systems allows to increase the level of their readiness and provide the additional protection from common cause failures.

3. DESIGN SOLUTIONS ON ECONOMICS IMPROVEMENT

The solutions aimed at the efficiency economical indices have been investigated together with the development and substantiation of solutions on safety increase in NV NPP-2 project. The design solutions for realisation of these targets can be divided into two groups:

- Design solutions aimed at cost decrease for NPP construction;
- Design solutions aimed at the increase of reliability of power production and cost decrease for operation.

3.1. Cost decrease for NPP construction

The calculations of basic economical parameters for new projects of NPP with WWER-1000 V-428 developed by the institute “Atomenergoproekt” for Tanvan NPP in Chine, V-392 for NV NPP in Russia and the reactor WWER-1500, the project of which will be developed after 2000 are given in Table III. As we can see, the project with V-392 exceeds considerably the economical parameters of the project V-348. In particularly specific capital investments in the construction under the project NV NPP-2 are 1.4 times less than for the project of NPP with the reactor V-428 and 1.6 times less than for NPP with V-320. The calculated frequency values of core damage for the project NV NPP-2 (4.8 E-8/reactor per year) are about 100 times less than for the project of Tanvan NPP (5.0 E-6 /reactor per year).

The main effect of economical indices improvement in the project NV NPP-2 has been achieved because of the use of combination principle that is the combination of safety and normal operation functions.

Thus the use of probabilistic system of core emergency cooling, combining in itself the function of coolant maintenance during high and low pressure in an active core, a function of sprinkling system and a function of heat removal from spent fuel allowed to exclude four channels of the core HP emergency cooling system, four channels of sprinkling system as well as the heat removal systems from fuelling 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 V-320 in the project of NPP with V-428.

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.

TABLE III. EFFICIENCY OF THE PROJECT

Item	**Parameters**	**WWER-1000 (V-320)**	**WWER-1000 (V-392)**	**WWER-1000 (V-428)**	**WWER-1500**
1.	Unit power, MW	1000	1068	1060	1470
2.	Life-time (year)	30	40	40	50
3.	Power service consumption, %	6.11	5.8	6.1	5.7
4.	Specific physical parameters of the projects				
4.1	Specific construction volumes (m^3/kW)		530	520	478
4.2	Specific consumption of reinforced concrete (m^3/MW)		95.5	129.9	81.9
4.3.	Metal consumption of pipelines t/MW including stainless steel t/MW		3.6 0.6	3.8 1.25	3.1 0.9
4.4.	Material consumption of electric equipment t/MW		2.6	3.74	2.5
4.5.	Length of power cables: more than 1000 V (km/MW)		0.025	0.032	0.016
4.6.	Length of power cables: less than 1000 V (km/MW		0.8	0.89	0.7
4.7.	Length of test cables: (km/MW)		2.0	2.31	1.76
5.	Capital investments in the main building, total: million rubles including: § Constructional works § Assembly § Equipment § Other		1060 110.4 118	1446.7 130.9 191.8	1112.4 120.8 125.4
6.	Specific capital investments in the industrial construction, including main building rubl/KW	1411	920.1 500	1297.6 677.7	826.9 378.3
7.	Cost of power production, kopecks/kW	3.43	2.11	2.18	1.62

Annex 9

Development of new nuclear power plant in Argentina

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Abstract. Argentina has started the design of its own nuclear power, CAREM. The CAREM is an indirect cycle reactor with some distinctive features that greatly simplify the reactor and also contribute to a higher level of safety: integrated primary cooling system, primary cooling by natural circulation, self-pressurised primary system and safety systems relying on passive features.

1. INTRODUCTION

Argentine Nuclear Development started in early fifties. Initially the activities of the Comisión Nacional de Energía Atómica (CNEA) of Argentina were oriented to research in nuclear physics, radiochemical studies, material science among others subjects. In 1957, the CNEA decided to build a Research Reactor. The RA-1 was the first nuclear reactor to be put in service in South America. Since then, Argentina has designed and constructed several Research Reactors in Argentina and another countries, and at the present competes with foreign developed countries as supplier of this technology.

In 1964, CNEA initiated the feasibility study for the construction of Atucha I Nuclear Power Plant (CNA I) which would be the first nuclear power plant in Argentina and Latin America designed for electric power generation. In 1967 entrusted its design and construction to Siemens. The construction began in June 1968 and the commercial operation started in June 1974. The station contains a reactor of the pressure vessel type and it is heavy water moderated and cooled being of the PHWR type; it is periodically refueled on power. CNA I 's original design considered only natural uranium as fuel, being its electric power of 340 MWe. The station suffered two essential modifications that improved its performance:

– In 1977 the electric power was increased to 357 Mwe;
– Since 1995 a progressive loading with slightly enriched uranium (0.85 wt%) began, so that at present the core contains not only natural uranium fuel elements but also slightly enriched ones.

In 1967, CNEA initiated the feasibility study for the construction of Embalse Nuclear Power Plant (CNE) and in 1973 signed a contract with Atomic Energy of Canada Limited (AECL) and Societa Italiani Impianti P.A. (IT) for a 600 MWe CANDU–PHW (pressurized heavy water) type nuclear power plant. The construction of the station began in May 1974 and the commercial operation started in January 1984.

On the other hand, Argentina started the design of its own nuclear power plant, CAREM. The CAREM concept was first presented in March 1984 in Lima, Peru, during the IAEA conference on small and medium size reactor. CAREM design criteria or similar ones have since been adopted by other plant designers, thus originating a new generation of reactor design, of which the CAREM was, chronologically, one of the first. The Argentinean CAREM project, which is jointly developed by CNEA and INVAP, consists on the development, design and construction of an advanced, simple and small Nuclear Power Plant (NPP). The first step of this project is the construction of the prototype of about 27 MWe. This project allows Argentina to sustain activities in the nuclear power plant design area, assuring the availability of updated technology in the mid-term. This implies working with technology acquired in Research Reactors design, construction and operation, and Pressurized Heavy Water Reactors (PHWR) Nuclear Power Plant operation as well as developing advanced design solutions.

CAREM is an indirect cycle reactor with some distinctive features that greatly simplify the reactor and also contribute to a high level of safety:

– Integrated primary cooling system;
– Primary cooling by natural circulation;
– Self-pressurised;
– Safety systems relying on passive features.

2. TECHNICAL DESCRIPTION

2.1. Primary system

The CAREM reactor pressure vessel (RPV) contains the core, steam generators, the whole primary coolant and the absorber rods drive mechanisms (figure 1). The RPV diameter is about 3.2 m and the overall length is about 11 m.

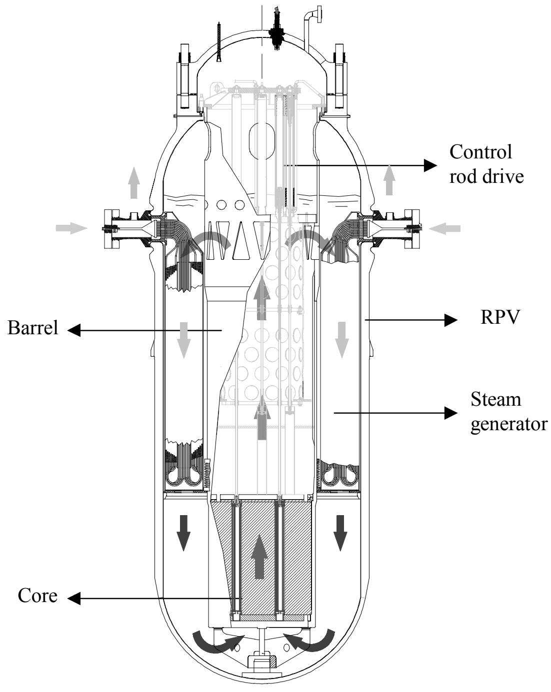


FIG. 1. Primary cooling system.

The Core of the prototype (figure-2) has 61 Fuel Assemblies (FA) of hexagonal cross section. Its components are typical of the PWR fuel assemblies. The fuel is UO_2 enriched at 1.8 and 3.1 wt%. An 8% weight of Gd_2O_3 is used as burnable poison to keep reactivity approximately constant along the fuel cycle. Chemical shim is not used for reactivity control during normal operation. Fuel cycle can be tailored to customer requirements, with a reference design of 330 full-power days and 50% of core replacement.

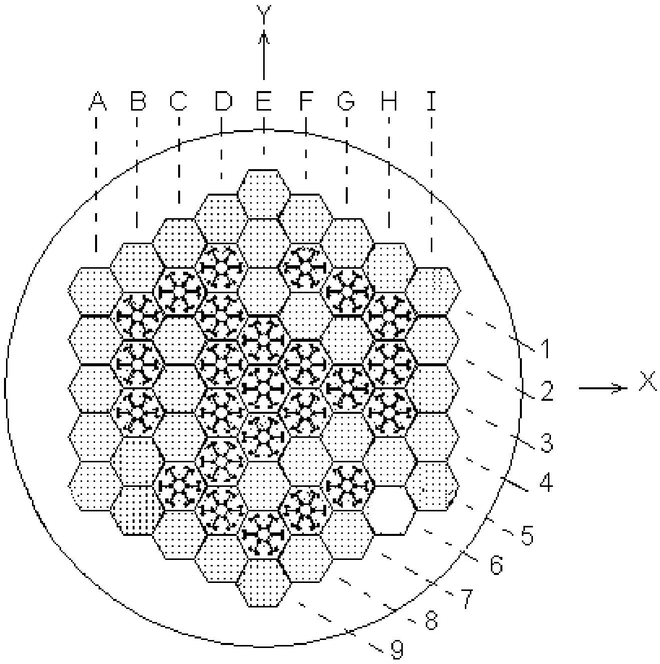


FIG. 2. Core cross section.

The core has 25 Absorbing Elements (AE). Each 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, at 18 positions in the FA not occupied by fuel rods. 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).

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). They are used to transfer heat from the primary to the secondary circuit, producing dry steam at 4.7 MPa, with 30°C of superheating (figure-3).

The location of the steam generators above the core produces natural circulation in the primary circuit. The secondary system circulates upwards within the tubes, while the primary does so 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.

In order to achieve a rather uniform pressure-loss and superheating on the secondary side, the length of all tubes is equalised by changing the number of tubes per coil layer. Thus, the outer coil layers will hold a larger number of tubes than the inner ones. Due to safety reasons, steam generators are designed to withstand the primary pressure without pressure in the secondary side. The steam system is designed to withstand primary pressure up to isolation valves (including the steam outlet / water inlet headers) for the case of SG tube brake The natural circulation of the 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.

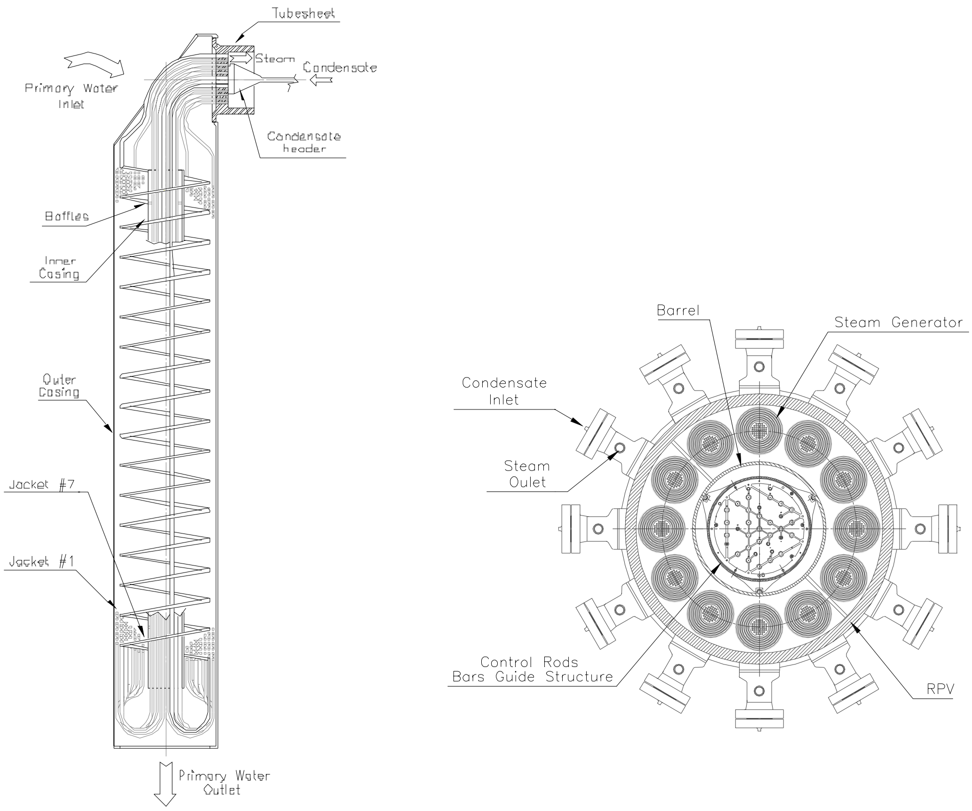


FIG.-3. Steam generators.

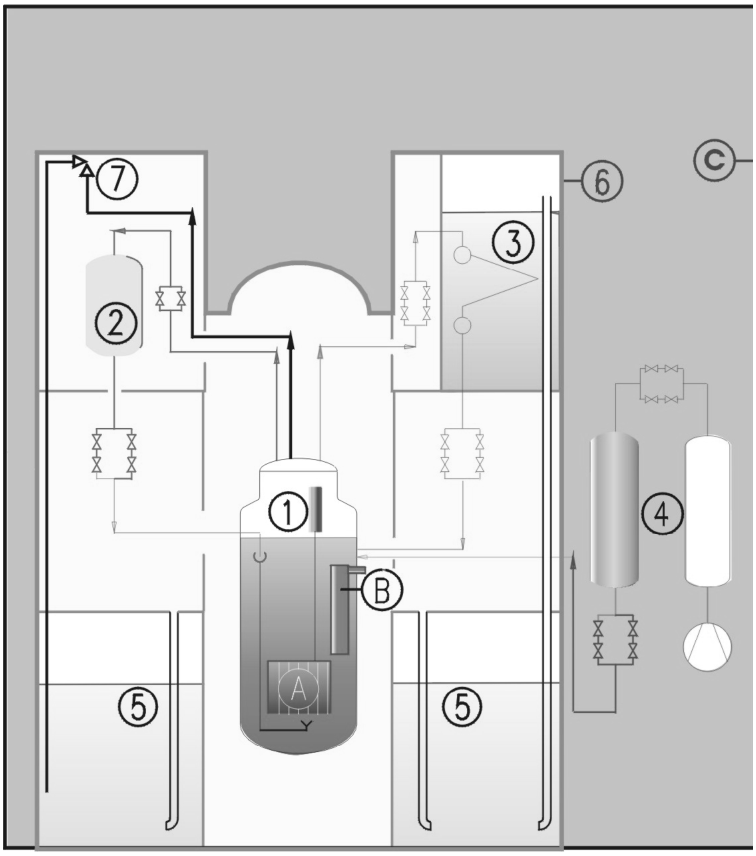
Due to the self-pressurising of the RPV (steam dome) the system keeps the pressure very close to the saturation pressure. At all the operating conditions this has proved to be sufficient to guarantee a remarkable stability of the RPV pressure response. The control system is capable of keeping the reactor 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-pressurisation features make this behaviour possible with minimum control rod motion.

In summary, the reactor has an excellent behaviour under operational transients.

2.2. Safety system

CAREM safety systems are based on passive features and must guarantee no need of active actions to mitigate the accidents during a long period. They are duplicated to fulfill the redundancy criteria. The shutdown system should be diversified to fulfill regulatory requirements.

The First Shutdown System (FSS) is designed to shut down the core, when abnormal or deviated from normal situations occur, and to maintain the core sub-critical during all shutdown states. This function is achieved by dropping a total of 25 neutron-absorbing elements into the core by the action of gravity.



SAFETY SYSTEMS

1- First shutdown system
2- Second shutdown system
3- Residual heat removal system
4- Emergency injection system
6- Containment
7- Relief valves

REFERENCES

A- Core
B- Steam generator
C-Secondary containment
5- Suppression pool

FIG. 4. Safety systems.

Hydraulic Control Rods Drives (CRD) avoid the use of mechanical shafts passing through, or the extension of the primary pressure boundary, and thus eliminates possibilities of big Loss of Coolant Accidents (LOCA) since the whole device is located inside the RPV. Their design is an important development in the CAREM concept. Six out of twenty-five CRD (simplified operating diagram are shown in figure-5) 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. The CRD of the Adjust and Control System is a 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.

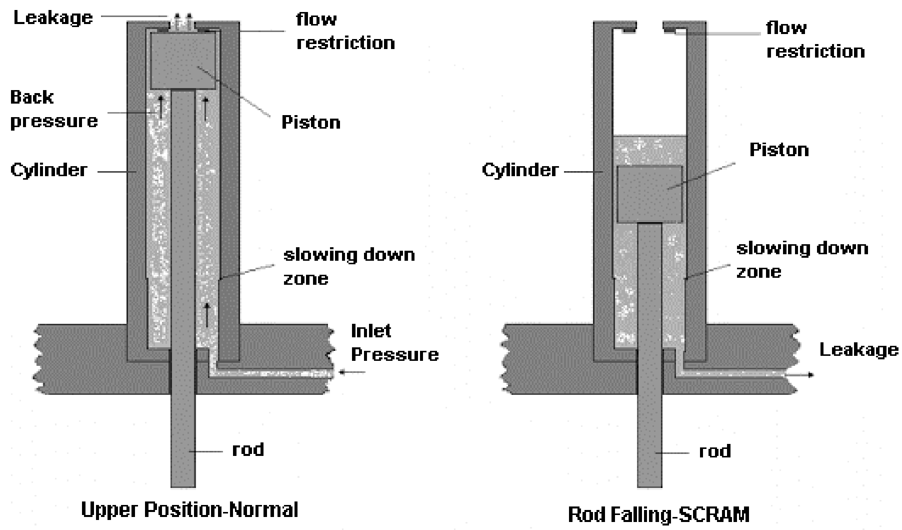


FIG. 5. Simplified operating diagram of hydraulic control rod drive. (Fast Shutdown System)

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 of 2 m^3 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 via piping 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 of 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 is condensed 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 (EIS) prevents core exposure in case of LOCA. In the event of such accident, the primary system is depressurised with the help of the emergency condensers to less than 1.5 MPa, with the water level over the top of the core. At 1.5 MPa a low pressure water injection system comes into operation. The system consists of two tanks with borated water connected to the RPV. The tanks are pressurised to 2.1 MPa, thus when during a LOCA the pressure in the reactor vessel reaches 1.5 MPa, the rupture disks break and the flooding of the RPV starts.

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.

For CAREM-25 accident analysis a nodalization of the primary circuit including SG was developed for RELAP5 and RETRAN02 codes. A simplified two zones non-equilibrium model was developed to calculate long term reactor behavior. RHRS and SSS were also modeled. Steam condensation on the absorber rods drive system and on RPV wall was implemented through boundary conditions. The reactor steady state at full power was calculated. The results agree quite well with design values, and this condition was used as reference for the accident analysis. Several initiating events were considered for the accident analysis. They were grouped into Reactivity Insertion, Loss of Heat Sink (LOHS) and LOCA. As there are no primary pumps Total Loss of Flow Accident (LOFA) is not applicable in this case.

A reactivity insertion accident in CAREM core can be produced by different initiating events: a cold water injection into the RPV, a secondary side steam line break and a failure in the absorbing rods drive system. The present work analyses inadvertent control rod withdraws transients. Results of the accident simulations with actuation of FSS show that safety margins (DNBR and CPR) are well above the acceptable minimum values.
The behavior of CAREM reactor and the Residual Heat Removal System to mitigate a loss of heat sink accident was also analyzed in the present work. RHRS design requirements to be fulfill for this accidental sequence are:

– *Short-term:* primary circuit pressure must remain below safety valves opening set point and condensers must not flood in order to avoid instabilities;
– *Long-term*: to reach hot-shutdown condition (primary circuit pressure below 2.3 MPa).

Short-term reactor behavior was simulated using RELAP5 with a detailed nodalization of the primary circuit and RHRS assuming different engineering factors. Long term performance was simulated with a simple and conservative model, assuming a saturated primary circuit. This condition is expected during RHRS operation. Results show that the requirements are verified and the reactor reaches hot-shutdown in approximately 35 hrs in a safe condition.

Finally, the CAREM-25 reactor response to LOCA was analyzed. A parametric study considering several break diameters (½”, ¾”, 1”, 1 ½” and 2”) in the steam zone of the RPV was performed. For each accidental sequence, the successful operation of one of the safety systems redundancy was modeled. A total Steam Generator feed-water loss and Chemical and Volume Control System (CVCS) unavailability are postulated when SCRAM occurs for a conservative calculation. Maximum loss of coolant flow, reactor power, safety systems trip time and core uncovery time were analyzed. The period analyzed shows that there is not an early core uncovery and there is no need of a high-pressure injection. The reactor remains cooled during that period.

As a general conclusion after the accident analysis, 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 severe accidents.

2.3. Plant design

The CAREM nuclear island is placed inside a containment system, which includes a pressure suppression feature to contain the energy of the reactor and cooling systems, and to prevent a significant fission product release in the event of accidents.

The building surrounding the containment has been designed in several levels and it 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 and reactor auxiliary systems, the spent fuels pool and other related systems in one block. The plant building is divided in three main areas: control module, nuclear module and turbine module.

Finally, CAREM NPP has a standard steam cycle of simple design.

3. ADVANTAGES OF CAREM DESIGN

Technical and economical advantages are obtained with the CAREM design compared to the traditional design:

– No large LOCA has to be handled by the safety systems due to the absence of large diameter piping associated to the to primary system. The size of maximum possible break in the primary is 38 mm;
– The rod ejection accident has also eliminated due to the development of innovative hydraulic mechanism located completely inside the reactor pressure vessel. In addition, hydraulic control rod drive mechanism significantly cost down compered with the current PWR’s control rod drive mechanism;
– Large coolant inventory in the primary results in large thermal inertia and long response time in case of transients or accidents;
– Shielding requirements are reduced by the elimination of gamma sources of dispersed primary piping and parts;
– The large water volume between the core and the wall leads to a very low fast neutron dose over the RPV wall;
– Eliminating primary pumps and pressuriser results in lower costs, added safety, and advantages for maintenance and availability.

4. CONCLUSIONS

The CAREM project consists of the development, design and construction of the prototype of an advanced small nuclear power plant. The CAREM is an indirect cycle reactor with some distinctive features that greatly simplify the reactor and also contribute to a higher level of safety: integrated primary cooling system, self-pressurised, primary cooling by natural circulation, safety systems relying on passive features. Therefore, many technical and economical advantages are obtained with the CAREM design compared to the conventional designs.

Key thrusts in next generation CANDU

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Abstract. Current electricity markets and the competitiveness of other generation options such as CCGT have influenced the directions of future nuclear generation. The next generation CANDU has used its key characteristics as the basis to leap frog into a new design featuring improved economics, enhanced passive safety, enhanced operability and demonstrated fuel cycle flexibility. Many enabling technologies spinning of current CANDU design features are used in the next generation design. Some of these technologies have been developed in support of existing plants and near term designs while others will need to be developed and tested. This paper will discuss the key principles driving the next generation CANDU design and the fuel cycle flexibility of the CANDU system which provide synergism with the PWR fuel cycle.

1. THE PATH TO NEXT GENERATION CANDU

The Pressurized Heavy Water Reactor (PHWR) CANDU system is a mature technology that evolved from 55 years of nuclear technology development and 30 years of commercial operation in many countries. Today thirty (30) CANDU units are operating or under construction in seven (7) countries.

The first CANDU designs were originally predicated on optimal thermal neutron utilization to enable the use of natural uranium as a fuel. However, the CANDU system, like all high technology products, must evolve quickly to meet the new requirements of the 21st century power market. The next major step in this innovative development is called the Next Generation CANDU.

This "next generation" will retain all the characteristics of the present CANDU reactor, including high neutron economy, modular design, on-power fueling, passive safety, and simple fuel design. These characteristics enable a logical and systematic approach to advancing the design through an evolutionary process. In addition, some of these characteristics allow the technology to be applied to many conceivable advanced fueling strategy without having to change the basic concept.

Thus, the main principles established for future development:

a) retain CANDU characteristics,
b) ensure every component performs at its highest level,
c) simplify and eliminate,
d) maintain safety margins,
e) improve operability and maintainability, and
f) improve the efficiency of the process and resource use.

2. KEY THRUSTS OF NEXT GENERATION CANDU

Three key goals drive the development of the next generation of CANDU plants. These are:

- *Improved Economics (Capital and Operation):* Cost reductions will result from plant optimization and simplification using "enabling technologies" which increases efficiency without compromising safety or operating margins. A key aspect of plant optimization is to ensure that all components and systems are performing at peak performance.

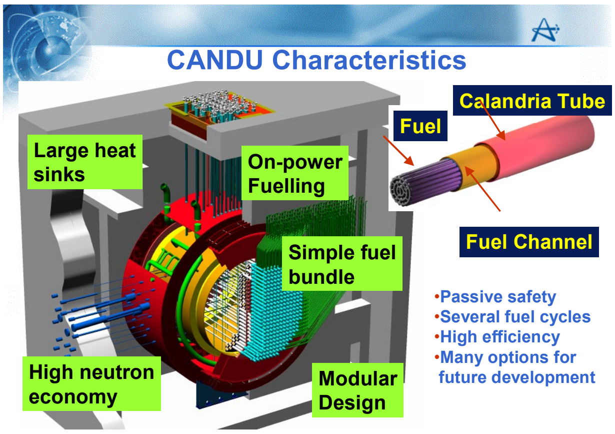

FIG. 1. CANDU characteristics.

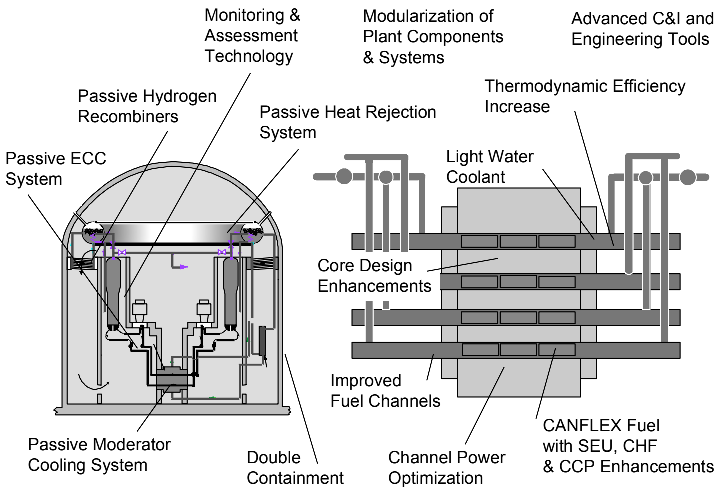

FIG. 2. Next generation CANDU enabling technologies.

- *Safety Enhancements:* The emphasis is on passive safety, which increases the reliability of safety systems while reducing design and operating complexity.

- *Enhanced Plant Operation:* The use of advanced technologies, such as "Smart CANDU" concepts to monitor and predict plant performance, will be implemented to maintain high capacity factors over the life of the plant.

To achieve these goals the next generation reactors will use new enabling technologies that are spin offs from the key characteristics of the CANDU design. Some of these enabling technologies have been developed for use in existing CANDU plants and the CANDU 9 design while other advancements will require extensive development and testing over the next few years.

2.1. Improved economics

The current CANDU reactors using natural uranium fuel are highly competitive with other commercial reactor systems. They are also competitive with fossil plants in many markets in terms of the lifetime unit energy costs, owing to the relatively low fuelling costs of nuclear power. However, it is recognized that many markets require energy systems to be competitive with low capital cost alternatives, such as combined cycle gas turbines (CCGT), even if the unit energy costs of nuclear power are lower. This is particularly true under market conditions where the long-term cost of electricity and price stability are less important than a short-term return on investment. Such market conditions would likely prevail in an open market where large centrally-controlled utilities no longer have a monopoly position. Therefore, to ensure competitiveness and diversity of supply in such markets, it is essential to seek major reductions in capital costs.

Overall, we believe that the cost of nuclear power plants must be reduced by at least 33% to meet the requirements of the coming decades. Such a cost reduction would significantly expand the market for nuclear power, particularly in emerging markets where the cost of capital is a major factor. This, in turn, would have a major effect on the reduction of environmental emissions (especially greenhouse gases) over the coming decades. In more developed markets, the goal is to provide an energy mix that would allow the continuing use of hydrocarbon resources without possible restrictions due to greenhouse gas emissions

A general methodology has been developed for meeting the economic target. Since the CANDU core design is highly flexible, the initial step is to optimize the core to ensure that the maximum power can be extracted for the resources used (i.e., heavy water, fissile material, coolant flow, etc.). Once this is done, the remainder of the plant can be resized or improved, since the core characteristics drive or are tightly coupled to many of the other costs in the overall system. Core optimization starts with the fuel, which can be enhanced to ensure that the optimum energy can be extracted from the fissile material. Next, each fuel channel must be optimized with respect to channel power output. Once the output of each channel has been optimized, the entire core can be improved to provide the highest output for the smallest volume (thus, for example, improving the power to heavy water ratio). The heat transport system (HTS) and turbine-generator are then optimized based on the total core power. Finally, process systems and components are examined in detail to ensure their “fit” with the enhanced core configuration.

Every component in the plant is being examined and evaluated against both the goals for the NG CANDU while preserving the CANDU characteristics discussed above. A key element of our strategy to improve the economics is to optimize or eliminate expensive components. For example, heavy water is an essential component of the CANDU PHWR and provides moderation for high neutron economy. However, for the Next Generation CANDU opportunities to eliminate the use of heavy water where it is not strictly required for moderation, such as the coolant system, are being examined. In a similar way, every component and system is being challenged, to ensure that it is only performing its highest-level function.

2.1.1. Fuel and fuel channel optimization

CANDU fuel, owing to its simple design and location in a fuel channel with a well-characterized flow, can be optimized by improving the distribution of coolant flow and heat generation throughout the bundle. The evolution of CANDU fuel has led to progressively higher performance by segmenting the fuel into smaller elements. The latest CANDU fuel design (CANFLEX), takes this evolution a step further by using different element diameters as well as increasing the segmentation. The result is a fuel bundle that produces the same thermal output as the current fuel design, but at 20% lower maximum linear element ratings. Operating margins have also been enhanced by optimizing the flow properties in the subchannels of the fuel to increase the critical heat flux limits and the critical channel powers. This optimization is the result of detailed understanding of subchannel flows and heat transfer; it is yet another manifestation of the time-proven methodology of parallel experimentation and mathematical modeling, which AECL has used for several decades to develop and evolve the CANDU system.

CANFLEX fuel can reach burnups that are approximately three times the current 37-element fuel bundle. The higher burnups can be achieved by switching from natural uranium to slightly enriched fuel (SEU). For example, using 0.9% SEU fuel would double the burnup compared to the natural uranium fuel. By adopting 1.2% SEU, the burnup could be almost tripled. Such burnups would reduce the volume of spent fuel by a considerable amount, as well as reducing load on the fuelling machines.

CANFLEX SEU fuel can also be used to improve the channel power output, owing to the improved CHF margins and lower linear element ratings. At enrichments of only about 1.5% (still well below the ≥3.5% enrichment used in LWRs), CANFLEX enables further optimization of the CANDU core by enabling the use of light water as the HTS coolant. Such a core design would still retain the key physics advantages of the CANDU reactor. Cost analyses to date show that elimination of heavy water from the HTS more than overcomes the cost of slightly increasing the enrichment. Not only is the total cost of heavy water in the plant reduced substantially, but also the cost of auxiliary systems for heavy water recovery and treatment. By using heavy water only in the relatively low pressure and temperature moderator, then heavy water recovery and treatment systems can be reduced in size or eliminated. This reduces/eliminates the operating and maintenance costs of those support systems.

In addition to improving the core thermal power output, the efficiency of electricity production can be improved by increasing the temperature of the heat transport system (HTS) coolant. For the Next Generation CANDU design, we have targeted a thermal efficiency improvement to about 36% by increasing the temperature of the HTS coolant to 330^{o}C. To accommodate these conditions, we are developing a slightly thicker and more corrosion-resistant pressure tube, with improved fuel channel components. At the same time, by using SEU and optimizing the core configuration, we can still retain the high neutron efficiency of the core despite the increase in pressure tube thickness.

2.1.2. Heavy water reduction

There are two approaches for heavy water reduction. First, as discussed above, by enhancing the power output from the core, fewer channels are needed for the same total power output. This, in turn, reduces the calandria size and the heavy water volume. An illustration of this is given in Figure 3, which compares calandria size for the CANDU 6 with the 600 MWe next generation CANDU concept discussed above. The reduction in size reduces the heavy water moderator requirement by a factor of 2.5. The second approach to reducing heavy water is the use of light water coolant in the heat transport system (HTS) and optimization of the channel pitch. This, combined with the reduction in moderator size, reduces the requirement for heavy water by more than a factor of 4. The absence of heavy water from the high pressure HTS reduces the load on heavy water systems, and reduces both capital and operating cost.

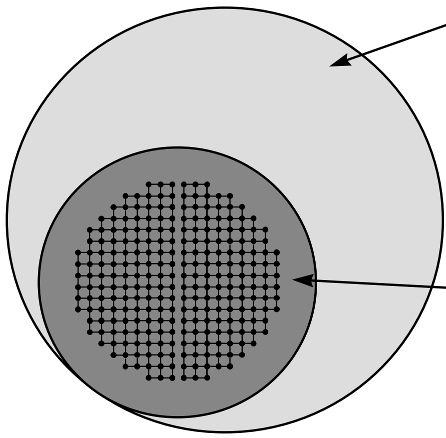

FIG. 3. Calandria reduction by core optimization.

The use of light water in the HTS also greatly simplifies the emergency core cooling/HTS interface. By trading off channel pitch and fuel enrichment, the coolant void reactivity could be reduced to any value desired (including negative). It is important to note that such changes would not affect the overall neutron economy of the CANDU reactor, and the use of advanced fuel cycles (such as the Direct Use of PWR Fuel in CANDU (DUPIC)) would not be restricted.

2.2. Enhancements in passive safety

CANDU reactors are unique in that a loss of coolant and loss of emergency core coolant does not lead automatically to severe fuel damage. The reason is the presence of the moderator, which can effectively and passively remove heat from the fuel. Over the years, we have improved the heat transfer from the fuel to the moderator under accident conditions by making small modifications to the fuel channel design. In the future, we intend to take this passive concept a step further by using thermosyphoning to remove heat from the moderator. The heat is then deposited in a large water reservoir, such as the reserve water tank used for the CANDU 9. The concept has been assessed and tested in large-scale laboratory tests for simple configurations. A similar system could also be used for normal operation, and the heat recovery used for feedwater heating to further improve the thermal performance of the plant.

Safety enhancements can also lead to reductions in complexity and cost. The replacement of valves with rupture discs in the emergency core cooling system (ECCS) of CANDU 9 design that rupture when the pressure in the HTS drops below a prescribed level have resulted in ECC reliability improvement. Such reduction in the number of valves has also reduced both capital and maintenance costs.

In addition to enhancing the various heat sinks and cooling systems, AECL is also developing other passive safety technology. A prime example of this is a passive autocatalytic recombiner, which is used to reduce hydrogen that could be released to containment. The recombiner works under cold, wet conditions by employing a proprietary wet-proof catalyst that does not require any active systems or power to operate. The recombiner maintains hydrogen concentrations below the combustion limit for some postulated accident conditions. With this recombiner design, the higher the hydrogen concentration, the more effectively the recombiner works owing to the increase in flow through the device as more heat is generated by the hydrogen/air reaction. This technology is replacing conventional igniters in AECL's future plants.

2.3. Improved operability and OM&A costs

For the Next Generation CANDU, a design goal is to enhance operating margins. Examples of improvements will include:

- Improved Regional Overpower Protection margins
- Reduced coolant void reactivity
- Lower fuel element ratings and greater margin for higher burnup
- Increased fuel critical heat flux margins
- Increased margin in pressure tube end-of-life properties.

These enhancements will be further developed and characterized in our development programs over the next few years. They are expected to have a strong impact on both plant economics and on lifetime capacity factors.

AECL is also developing the "Smart CANDU" suite of technologies, which will greatly enhance operability over the life of the plant. The "Smart CANDU" concept uses a combination of diagnostic probes, historical data bases, state-of-the-art codes, and advanced information technology to provide operators with both the current and future status of the critical systems, structures, and components in the plant. For plant construction, the advantage of such technology is that equipment will not have to be over-specified to ensure that it operates within its design envelope over the life of the plant.

One of these technologies is called ChemAND (Chemistry Analysis and Diagnostics). ChemAND is a general plant chemistry information tool that features automated monitoring, alarming, diagnostics, prediction, and online execution of analysis codes.

The next technology in this series, ComAND (Component Analysis and Diagnostics), will provide similar information on the critical plant components. In the future, we also plan to address thermal margins by incorporating system health monitors to measure heat transfer, flow, and other parameters affecting thermal performance. Such a system will allow plant optimization as well as avoiding potential de-rating due to premature aging effects.

These technologies would enable new business models for plant operation. A future operator may wish, for example, to draw on external expertise to monitor the plant, and to recommend maintenance requirements and operating conditions. Such an operating model would make it easier to adopt nuclear energy without the expense and time of having to create and maintain all the expertise in-house. It could also lead to more risk/benefit sharing arrangements, whereby the vendor could take on more responsibility for economic operation of the plant.

2.4. Fuel cycle flexibility

Countries with nuclear plants that wish to retain self reliance and energy independence need to explore the existing synergy between PHWR fuel cycle and that of the PWR.

The CANDU reactor is unique in that several viable fuel cycles are possible using both fissile and fertile fuel. All present and future CANDU designs will continue to accommodate these fuel cycles by maintaining high neutron efficiency, simple fuel bundle design, and on-power fueling. Even using SEU, neutron economy will still be optimized and CANDU's ability to burn a wide variety of fuels will be retained.

CANDU advanced fuel cycles (i.e., beyond the use of natural uranium and SEU) fall into two main categories. The first is the recycle of existing fissile material, such as spent PWR fuel. Spent PWR fuel represents a valuable resource for neutron efficient CANDUs, since it contains 1.5% fissile Pu and U. By reconstituting the spent fuel, using a relatively simple and proliferation-resistant dry process without Pu-U separation, to convert the fuel material into CANDU fuel pellets (DUPIC = Direct Use of PWR fuel In CANDU), an additional ~ 15000 MWd/t could be extracted from the spent fuel. In addition to DUPIC, there are a number of other fuel cycles using PWR fuel reprocessing wastes that are viable for CANDU, including the recycling of recovered uranium, plutonium, and even fissile and fertile actinides.

The second category of advanced fuel cycles concerns the extension of fissile material well into the future. For the CANDU, the development of new advanced (and expensive) technologies, such as Liquid Metal Reactors (LMRs), is not required to secure a long-term source of fissile material. One advanced fuel cycle available to CANDU reactors now is the option to burn thorium fuel. This would extend CANDU HWR applicability for the foreseeable future without having to develop a new type of reactor. A number of thorium fuel cycles are possible, including once-through cycles that do not involve fuel processing. CANDU reactors can also support LMR-based cycles, since one LMR could produce sufficient fissile material to fuel up to nine CANDU reactors. This is in contrast to the 1:1 ratio if the LMR/PWR cycle were adopted. Since the LMR will likely be an expensive commercial reactor, the CANDU/LMR synergism would be a more cost-effective option. The main option for thorium cycles that has been considered up to now is recycling ^{233}U from the spent fuel. An alternate approach is to adopt a once-through thorium cycle that does not depend on recycling of fissile ^{233}U. A "mixed bundle" approach, where elements of thorium and enriched uranium are contained in the same bundle, is the most attractive option from the perspective of fuel management and reactor control.

In the context of protection against proliferation, the CANDU is subject to International Safeguards and offers no diversion disadvantages compared to other current reactor designs. For future fuel cycles, thorium has an added advantage in that production of the fissile isotope ^{233}U unavoidably results in production of other uranium isotopes that make the fuel effectively unusable in nuclear weapons. The predominant reasons that thorium has not been used more widely to date is the fact that the ore must be 'enriched' with either ^{235}U or plutonium to start this fuel cycle, and the overwhelming advantage of experience with uranium fuels. However, at the present time there is an excess of separated fissile isotopes in the world, some of which could be used for introduction and development of thorium-based fuels for the future.

2.5. Constructability

The Next Generation CANDU will also draw heavily on past design experience with previous CANDU reactors. In recent years, considerable attention was paid to plant layout, materials, and constructability. A good example of this is the Qinshan project in China, where partial open-top construction techniques using heavy lift cranes are helping AECL and our partners to meet an ambitious construction schedule. This approach has been further advanced for the CANDU 9 design, where extensive modularization of components, optimal plant layout, and open top construction will lead to even shorter construction times. Similar modularization will be designed into the Next Generation CANDU, and, as a stretch target, we have established a goal of 36 months from first containment concrete to in-service.

These advancements along with the new features discussed above will also improve plant construction. As an example, a smaller calandria with a reduced number of fuel channels would allow a prefabricated calandria to be lifted into position with the fuel channels and reactor face feeder runs already installed. The smaller calandria size would also facilitate the installation of an integral stainless steel shield tank. The use of light water as the HTS coolant means that commissioning will be much simpler and faster – for example, there would be no need to test the hydraulics with light water, and then drain the system and refill with heavy water.

SUMMARY

The current competition from the Combined Cycle Gas Turbine (CCGT) and the emerging deregulated electricity market have defined the path for future development of nuclear generation. The PHWR-CANDU design is driven by the same market environments both in Canada and abroad. Improved economics, enhanced passive safety features and optimized constructability are some of the thrusts driving the development of the next generation CANDU.

New enabling technologies spun off from current CANDU features will form the basis of the next generation design; some have been developed for use in existing and new CANDU plants and others will require development and testing in the next few years.

ACKNOWLEDGEMENTS

Many AECL staff are aggressively pursuing the advanced knowledge in both the engineering and R&D required to take CANDU technology to the next stage of development. The degree of enthusiasm and creativity of these colleagues, who are too numerous to list here, is highly gratifying to the authors, and this paper is dedicated to their efforts.

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What it would take to order new nuclear plants — Japanese perspective

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Abstract. In most of the OECD countries, new nuclear capacity addition has been limited for the last one or two decades due mostly to the overcapacity or consideration of financial risk of capital-intensive nuclear investment. Japanese utilities have a dozen of new nuclear plants in a various stages of planning, licensing and construction. This is due to time-delayed demand and supply situation, a concerted effort to comply with the environmental agenda, and diversification incentives by regional Utilities and others. Beyond this stage, as Utility business deregulation progresses, new nuclear plant orders would depend on fundamental conditions such as the growth in electricity demand, competitiveness of nuclear power generating costs, and confidence in the Utility management of no stranded costs. Supporting institutional mechanisms such as environmental externality and the effort to cultivate confidence in the public for waste management and safety also help. This paper further discusses associated strategies to satisfy the fundamental conditions. This will range from strategies for replacement, technology development, and institutional arrangement to changes in Utility/Industry's structure & business practices.

1. INTRODUCTION

Deregulation of the electricity market has a fundamental potential to alter Utility corporate structure and business practices but it may also alter the power generating sources portfolio through competition in the electricity market. A capital-intensive nuclear power is prone to be considered as bearing such high financial risk that the investment may not be recovered from the competitive market. The conceived impediment to new nuclear plant installation stems not only from economics but political and regulatory instability and also public willingness. Against this background, concerted effort by IAEA membership countries for better use of nuclear power for the benefit and welfare of the public is deemed necessary.

2. NEW PLANT ORDERS IN THE WORLD AND JAPAN

2.1. The historical trend of the world's nuclear power plant orders

The historical trend of the world's nuclear power plant installation shows (Fig. 1):

a) The rapid growth and decline in the 70's and 80's among countries in Western Europe and North America;
b) Some delayed deployment in South and East Asian countries;
c) Active deployment in Japan after a decade of suspension.

The observed regional disparity in today’s environment, dorman in Western Europe and North America and active in South and East Asian countries, is due to such factors as new plant deployment in general (fossil or nuclear or other) in regions where electricity demand growth is visible (Fig. 2) and may also correlate to the advent of Utility business deregulation in the specific region and the domestic energy supply portfolio.

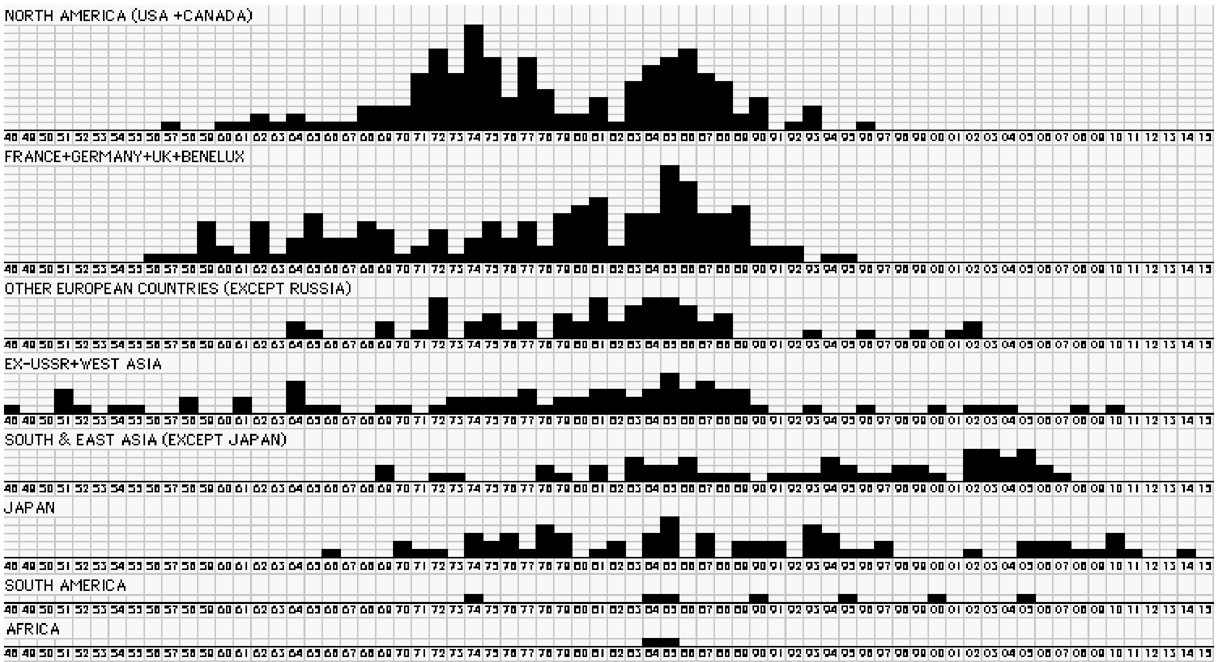
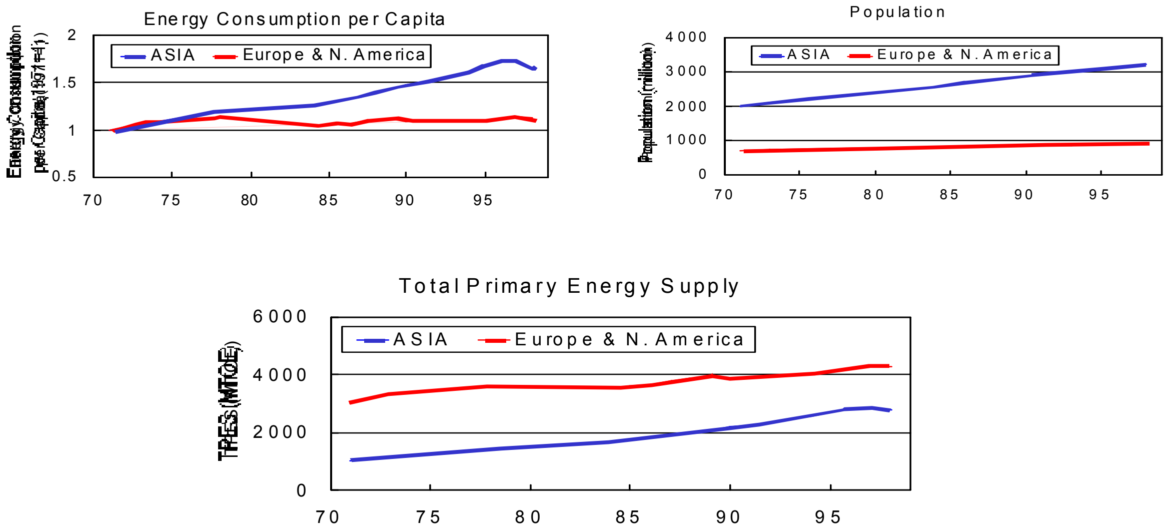


FIG. 1. The history of the world's nuclear power plant installation.



Source : IEA Energy statics & Balances of OECD Countries 97-98
I EA Energy statics & Balances of Non OECD Countries 97-98

FIG. 2. Comparison of regional growth in the world [1] [2]

In general it is observed that those countries with abundant domestic energy resources such as gas or coal tend to be less aggressively promoting the use of nuclear power than those with a scarcity of domestic resources. Japan, France and Korea typically belong to the latter group and would regard nuclear power as quasi-domestic resources based on the use of technological resources.

2.2. Current nuclear power plant deployment in Japan

2.2.1. New nuclear plant projects

Currently more than a dozen new nuclear plants (10 ABWRs, two other types of BWR, three PWRs) are in various stages of development ranging from planning, environmental surveying, licensing or construction. Most of these will start commercial operation before 2010 (Fig. 3). Of these, construction plans were authorized by the government for at least 6 units, and in the history of Japan, all but one nuclear plants (out of 51 units) were completed once construction plans were authorized.

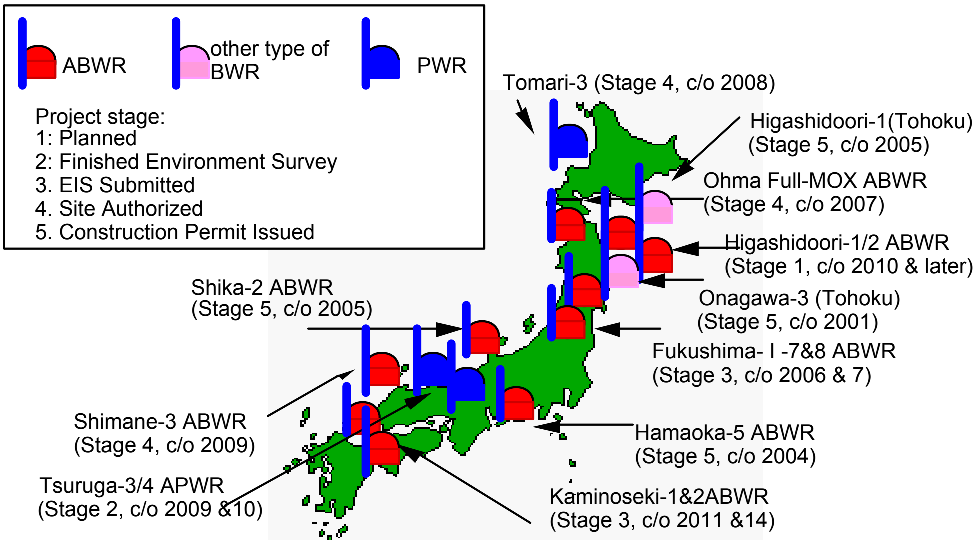


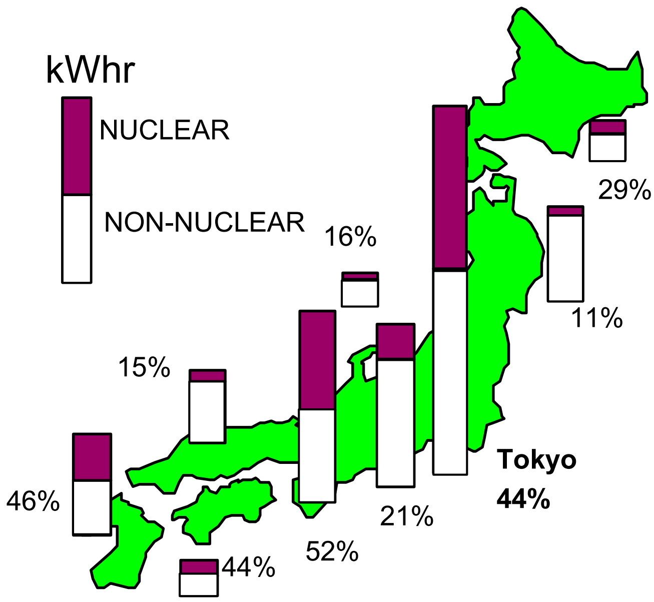
FIG. 3. New nuclear power plant projects.

2.2.2. Utilities incentives

The reasons for this active program in a country where electricity demand growth is relatively mild, especially in the wake of economic depression and Utility business deregulation can be explained as follows;

2.2.2.1. Diversification incentives by regional Utilities

Diversification of power generating sources has caused a strong drive for nuclear power among the Japanese Utilities which once depended on oil for around 80% of its electricity generation, and had experienced serious rate hikes in the wake of Arab Oil Embargo. Currently there is a observable disparity among the loosely-interconnected regional Utilities. Four regional Utilities (mostly large Utility) have a high percentage of electricity production from nuclear power (52%, 46%, 44%, 44%) and the remaining five have less than 30% (mostly 10-20%). It is not by coincidence that those Utilities with less nuclear electricity currently have active nuclear projects in advanced stages.



Kansai	52%
Kyusyu	46%
Tokyo	44%
Shikoku	44%
Hokkaido	29%
Chubu	21%
Hokuriku	16%
Chugoku	15%
Tohoku	11%

FIG. 4. Share of nuclear power of each regional Utility in Japan (FY 1999).

2.2.2.2. Economic perspective

The following estimation by the author for year 2020 is based upon respective Utility's publicly available financial reports that include FY1997-99 costs for power generations. The estimated relative economics of nuclear versus thermal power depends heavily on assumptions for fossil fuel price rise and nuclear fuel cycle cost. Waste disposal cost estimate is included for all types of waste. When asset depreciation progresses and fuel cycle and waste disposal costs are well controlled, nuclear electricity would remain competitive in year 2020 for those units already installed (The estimate included new units with ongoing stage or high probability of construction). This estimate (Fig. 5) is in line with the recent OECD report on nuclear power in deregulated environment [3] and is consistent with the information of current competitiveness of nuclear plants in the US. [4]

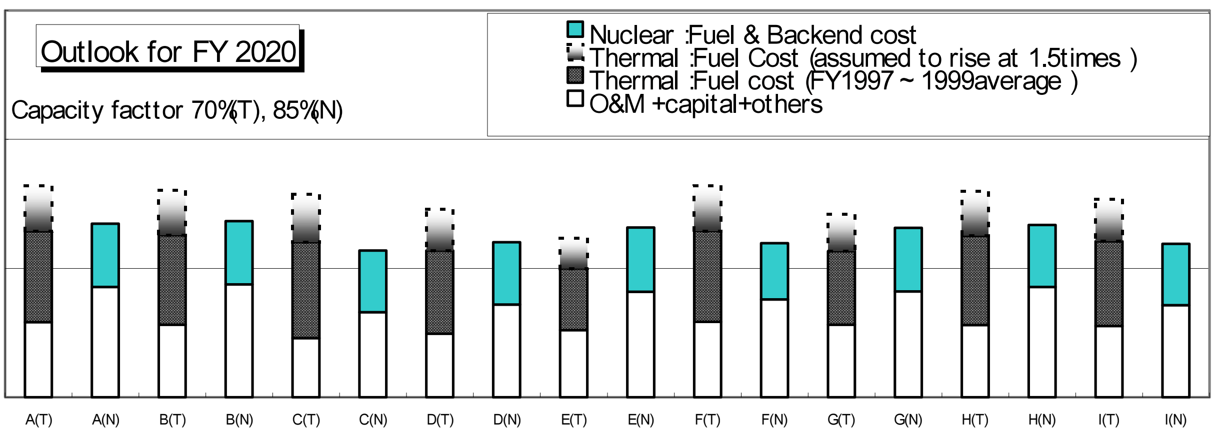


FIG. 5. Estimated economics of fossil/nuclear electricity in 2020.

2.2.2.3. Demand and supply situation

Some Japanese Utilities had seen a reduced reserved margin in the late 80's and the beginning half of 90's (Fig. 6), which motivated an active deployment program for all types of power generating sources. Nuclear power is not necessarily for peak load but its new deployment is affected by demand and supply situation.

New plant projects are now becoming a reality with a certain time delay ("latency effect" due to the long time required for consensus-building in the local community, environmental surveying and licensing.

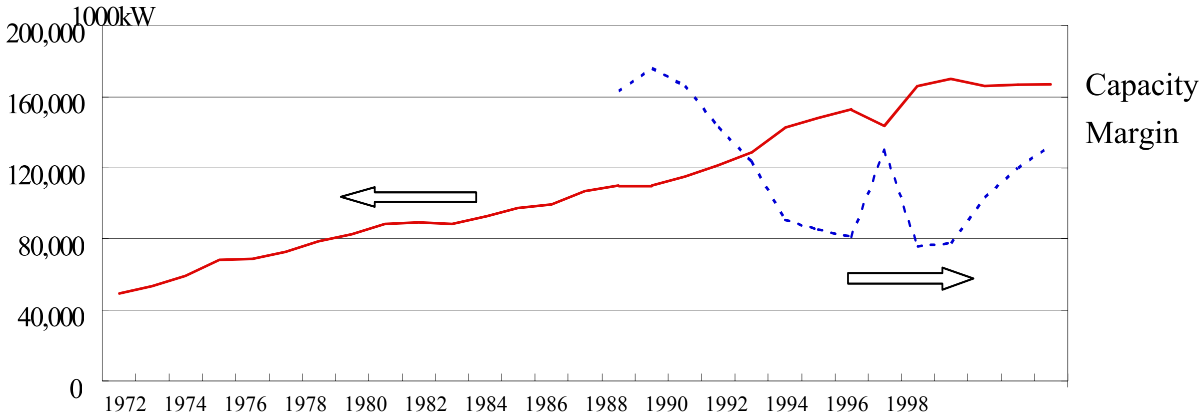


Fig. 6. Demand and supply situation and reserved margin (Japan).

However, a reserved margin is secured for most Utilities today, for instance, the addition of 7GWe to the grid over the last 4 years in a region where no kW increase is observed in the same period of time. Business deregulation starting in March 2000 has the potential to present a serious impediment for Utilities to invest in new capital-intensive nuclear projects in order to avoid financial risk.*

2.2.2.4. Environmental agenda

Utilities are expected to comply with the country-specific emission reduction targets set forth in COP3 (KYOTO, 1997), in which JAPAN promised to reduce global warming gas emission by 6% until year 2010 from 1990 level. [5]

The national plan to achieve this environmental agenda assumes the increased share of electricity from nuclear power by its capacity addition of 20GWe by 2010.

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* Utility business deregulation in Japan.
1st step: Amendment of Utility Business Law (Effective December 1, 1995)
Open the Wholesale Market to IPP & modify cost-plus rate making.
2nd step: Amendment of Utility Business Law (Effective March 1, 2000)
Open retail market to eligible customers (Contract w/ >20kV & >2000KW, 30% kWh)
Transmission lines remain as local monopoly. Access fee determined on the basis of forward-looking cost.
3rd step: Planned for three years later after review of status.

3. CONDITIONS FOR NEW NUCLEAR PLANT ORDERS

Conditions for new nuclear plant orders may vary depending on such factors as the type of ownership (privately owned or state owned) , the energy policy of the specific country of concern, and the level of Utility business deregulation.

However, they will generally rest on the following primary elements:

1) Demand growth and the reserved capacity in the network or connectable network;
2) Competitive power generating cost for new nuclear plants;
3) Supportive institutional mechanisms;
4) Public confidence in plant safety and waste management.

Although it is quite natural that new plant orders are a function of the prospect of electricity demand growth and the reserved capacity in the network or connectable networks, Utility management in deregulated countries tend to avoid investment until confidence is built that the subject investment is certain to be recovered and new plant construction is better than uprating of existing plants or purchase of operating plants from other utilities.

Since existing nuclear plants with asset depreciation well underway are competitive in the market, integrating nuclear power generating costs by using averaged generating costs over the generation of nuclear plants may help offset temporary stranded costs associated with the new nuclear plants.

Current market price of electricity does not account for costs that future generations will have to bear for environmental restorative actions or for incurred price hike due to energy supply security. Hence, it is expected that supporting institutional mechanisms are prepared in order for the decision-makers appropriately to take those factors into consideration when selecting from among alternative power generating sources.
Public confidence, especially nuclear plant safety and waste management, is a pre-requisite for the consensus building in the society. Credibility of people engaged in nuclear business would form the basis of this confidence.

4. ASSOCIATED STRATEGIES TO SATISFY THE CONDITIONS

This chapter raises some examples of strategies that may be considered by nuclear Utilities in order to bring the conceived plans for new nuclear plant to reality. These are relevant to the conditions for new orders discussed in the previous section.

4.1. Integrated nuclear power generating cost

Integrating nuclear power generating costs by using averaged power generating cost over all the generations of nuclear plants may help offset temporary stranded cost associated with new nuclear plants as shown on Fig. 7, because of the gain available through plant life extension*

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* Regulatory system for plant life extension and plant life management [6]
<Life @ start of operation> <institutional life extension mechanism>
Japan **Not specified** Operation of the next cycle after annual government inspection
+**PSR**(every 10 years) / **PLM** (every 10 years after age 25)
where, Japanese **PSR** (Periodic Safety Review) = (1) IPE (individual plant PSA) + (2) Review of operational Experiences + (3) Review against current licensing basis & new findings, and **PLM** = residual life assessment and planning for inspection/replacement for each plant w/age over 25.

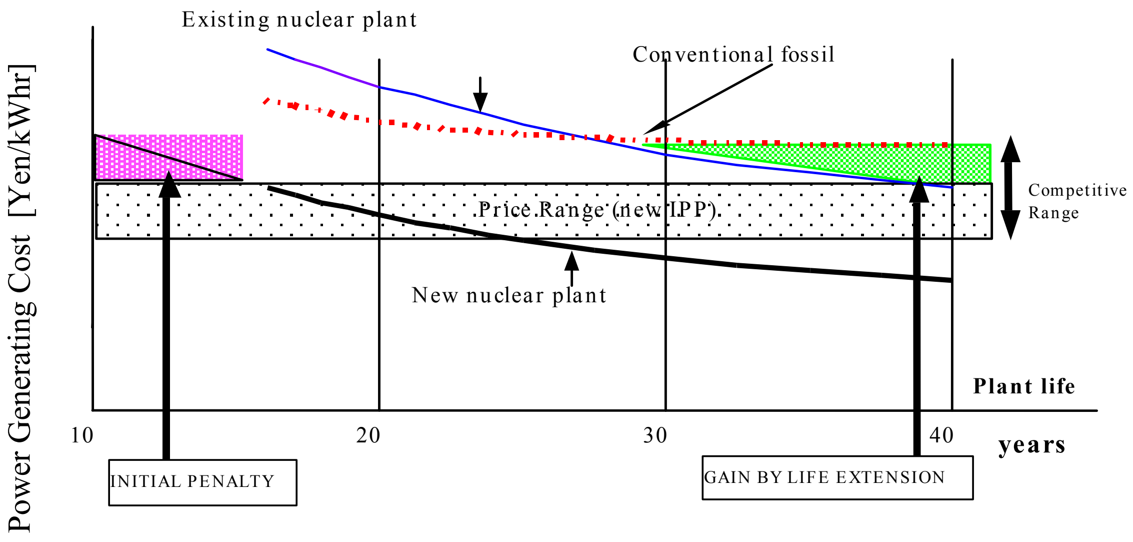

FIG. 7. Integrated nuclear power generating cost.

and power uprating experienced in various countries. Power uprating of existing nuclear power plants would be possible with small incremental cost as compared with the installation of new CCGT in terms of $/kW.

4.2. Levelized & controlled investment at the time of replacement

Japan, for instance, saw a sharp rise of investment in the 1970s. In case replacement of these units is to be planned on a simple programmatic basis that replace old units after pre-determined period expires, Utilities will face a sharp rise in investment. Consequently, we should levelize investment to avoid this situation and we will need a well defined program for replacement and new plant deployment that can be associated with technology development programs. (Fig. 8)

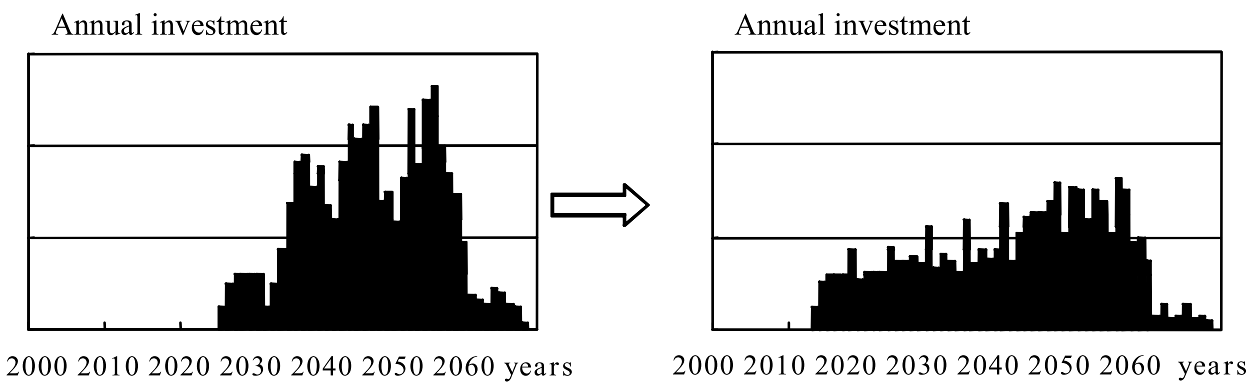

FIG. 8. Levelized investment.

For example, to control investment, a goal is set by TEPCO so that the total investment for new nuclear plant plus decommissioning costs for replaced units is less than the initial investment adjusted for escalation. This is made possible by fully utilizing existing infrastructure (land, harbor, transmission line etc) of existing plant sites and plant upscaling.

4.3. Incremental decrease in capital investment for new series

Standardization is a vitally important part of capital cost reduction. But it is often the case that the FOAK plant of new series is more expensive than the last unit of the previous series. This is relevant to the definition of the number of units for the recovery of T&D (Test &Development) and D&E (Design & Engineering) costs. The goal in TEPCO is that Δ1 is less than zero to smooth the transition and to control the T&D plus D&E costs in a reasonable range (Fig. 9). If the T&D plus D&E costs are to be recovered by the first four units of the series, it is shown that ABWR has achieved Δ 1 of almost zero as follows:

Δ 1 (ABWR(FOAK)-BWR5(Standardized))=-26$/kW
Δ 2 (ABWR(Standardized)-BWR5(Standardized))= -700$/kW due to standardization/scope split/others.
Naturally, TEPCO expects that Δ1 (ABWR-II(FOAK)-ABWR(Standardized)) would be less than zero, and
Δ 2 (ABWR-II(Standardized)-ABWR(Standardized)) would be minus 20-30% in terms of the magnitude of change.

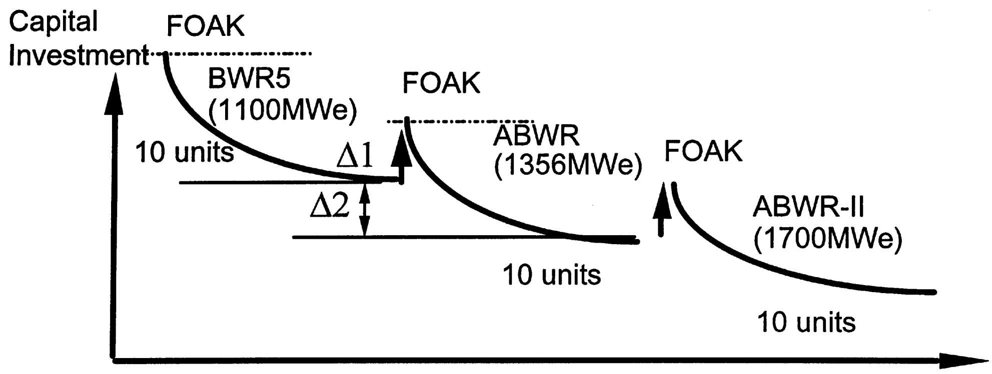

FIG. 9. Relative capital cost between series.

4.4. Control of Waste & Fuel (downstream) cost

As the investment level ($/kW) decreases by better technology and design and by high availability, waste & fuel (downstream) costs hold an increased share in nuclear power generating costs, especially in the case of countries with recycling policy. Without strict control of these costs, nuclear power will lose its competitive edge.

4.5. Change in Utility/Industry's structure & business practices

Re-organization of the nuclear industry including making alliances, M&A and expanding the business into international customer portfolios is visible in the shrinking nuclear market in OECD countries. On the part of Utilities, alliances for sharing resources among power stations or Utilities, M&A, purchase of operating units, and the transfer of good O&M practices to others are also prevailing. These changes in Utility and Industry's structure &

business practices would further enhance productivity of nuclear power and improve its competitiveness in the electricity market. Use of Information Technology to control the large volume of information for new plant design and construction will enable D&E cost reduction through concurrent engineering, will be beneficial in procurement in the e-market, and will help configuration management of plants after they start operation. In the case of TEPCO, some 200,000 design documents and 50m thick files of QA records are produced for each unit. The use of digital information control and project management tools such as ProjectNet is considered for pilot use for new nuclear facility in the next Fiscal Year in TEPCO. Also, a new form of collaboration by NSS vendors for T&D and D&E, "Virtual electronic consortium" is envisioned for the next generation BWR technology development.

4.6. Regulatory change

Modernization is expected, based on operational experiences and the advent of risk analysis methodology, for nuclear-related regulations in the area where the incremental cost increase associated with regulation does not positively correlate to the benefit of risk reduction. Utilities expect such regulatory changes as rated thermal power operation, extended operational cycle and use of risk information. It is estimated that the these, if permitted, would result in availability increase of more than 5%. Graded QA based on risk insight would enable equipment procurement from a large market and contribute to capital cost reduction.

4.7. Diversified options for future uncertainty

As Walt Patterson discusses in a book titled "Transforming Electricity" [7], two diverging paths into the future may exist for the future power generation. In fact, micro gas turbine and fuel cell technologies, although their share is limited, have a potential for energy supply that bypasses existing transportation/transmission networks. Decentralization would depend on technical achievement as well as economics that provide higher energy efficiency and versatility in the energy market.

4.8. Institutional scheme

As discussed previously (Chapter 3), the current market price of electricity do not account for the costs that future generations have to bear for environmental remedial actions or for incurred price hikes due to energy supply security.

This, combined with other factors, would raise a question on how to let both "energy policy agenda "and "market principle" stand. Consideration of such factors as environmental externalities will be necessary for decision-makers in selecting from among power generating sources. International organizations can help to establishing a defacto standard in external cost evaluation to be used in this process.

TABLE I. COMPARISON OF ENVIRONMENTAL EXTERNALITY [8]

		Coal & lignite	Oil	Gas	Nuclear	Biomass
Range (mECU/kWh)	Min	18	26	5	2.4	1
	Max	150	109	30	7.4	29

4.9. Public confidence in plant safety and waste management

Public perception of radiation and nuclear safety is a key in building a consensus for new plants. Of particular importance in today's environment are waste disposal (for which a significant progress is being made for HLW) and Spent Fuel Storage. Renewed public support for nuclear power is possible through providing the public with information to support their judgment, energy education in schools, accuracy in media reporting, and credibility of people engaged in the nuclear business.

Expected characteristics for the new nuclear plant design have been set as a candidate for replacement of existing nuclear plants. Neighbour friendly nature [Safety] as well as consumer friendliness [Economics] and user-friendliness constitute an essential part of the requirement for such designs. Neighbour friendliness requirement was defined in Japan for the design of next generation reactors in a way to comply with objectives of no evacuation and no land contamination. [9]

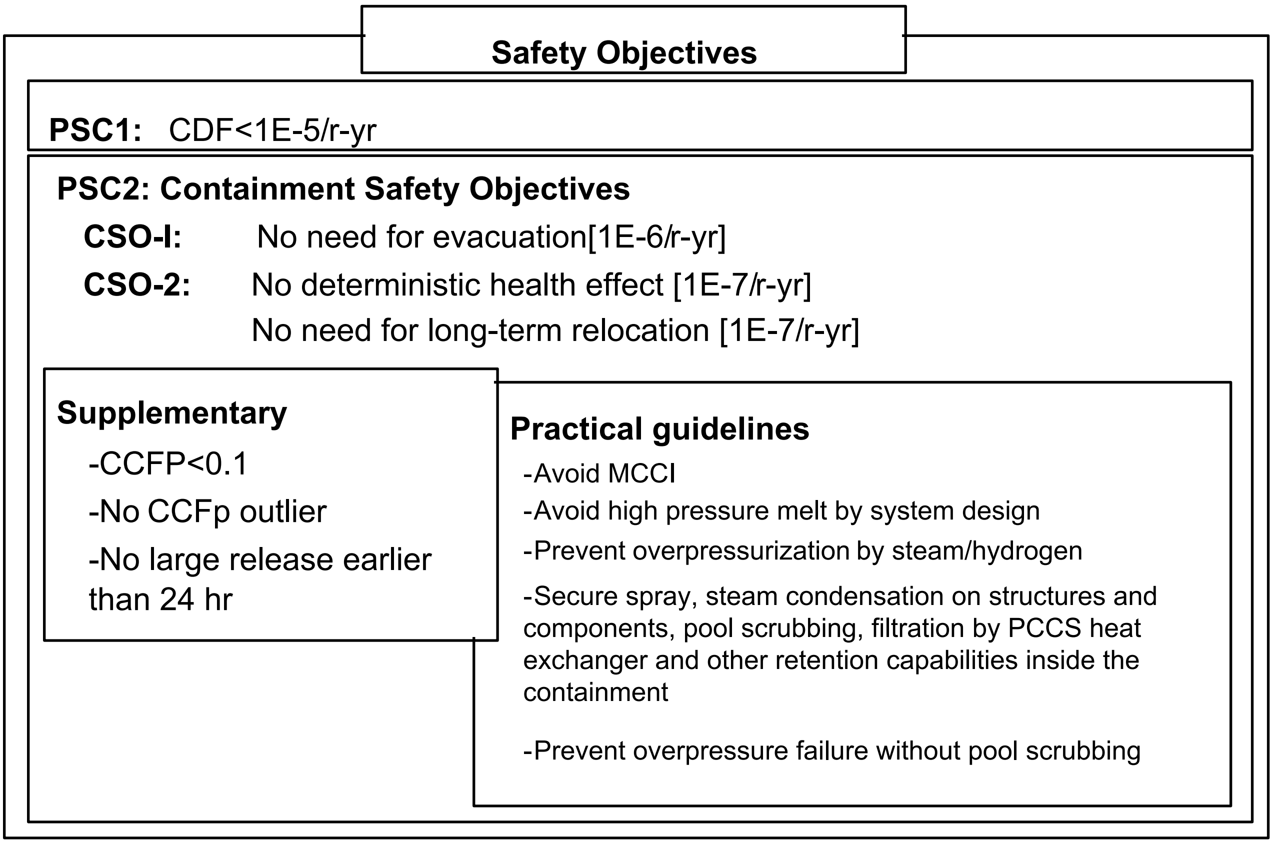

FIG. 10. Safety objectives for the next generation LWRs.

4.10. End-use approach

Deregulation of the Utility business provides Utilities with the opportunity to extend their services beyond just producing/transmitting/selling electricity. Looking at the advent of the technologies in the future, Utilities, as energy companies, can think about energy supply in the form of not only electricity and gas but also methanol or hydrogen. A simple calculation shows that if all the automobile is converted to EV and its electricity is supplied from nuclear power (equivalent to 20 ABWRs), we can save greenhouse gas emission by 20%. Use of nuclear power to supply alternative form of energy (heat, hydrogen, methanol) supply also have a potential to expand the horizon.

4.11. Technology development

Technological advance is at the root of any successful business.
Advanced technology development is actively being pursued in areas such as passive safety system, condition monitoring & inspection using micro-technology, new materials (cathode protection by photo-catalyst workable in Cherenkov radiation environment, shape memory alloy, self-diagnosis capabilities), simulation, and new structure (steel-sided wall & floor, seismic isolation, magnetic dumper) and so on for application for new plants.

5. CONCLUSIONS

Ongoing nuclear projects Japan have their background in diversification incentives, power generating costs perspective, demand and supply situations, and environmental agenda shared by the public and private sectors.

Conditions for new nuclear plant orders will be summarized as, a) demand growth and the reserved capacity in the network or connectable networks, b) competitive power generating costs for new nuclear plants, c) supporting institutional mechanism, and d) public confidence in waste management and safety.

Associated strategies to satisfy the above conditions would range from an investment strategy at the time of replacement, cost control target to Institutional scheme. Concerted efforts by Utility, Industry and the Government will smooth the way for revitalization of nuclear power including new plant orders.

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Cost reduction and safety design features of CNP1000

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Abstract. China will continue developing its nuclear power plant industry based on 3 units in operation and 8 units under construction. China National Nuclear Corporation (CNNC) recommends CNP1000 to Chinese government and customers as one choice of 1000MWe PWR Nuclear Power Plants (NPP) which will be built in near future. The feedback experiences of design, construction and operation for Qinshan phase II (QS-II) and Daya Bay Nuclear Power Plants are utilized in CNP1000 design. Self-reliance design, equipment manufacture localization, construction period reduction, standardization of reactor type, higher plant availability, enhanced management of NPP construction and so on are also considered to increase economic benefit and to reduce the cost of construction and operation. The measures such as reducing linear power density of core fuel rod, improving safety systems, using digital Instrumentation and Control (I&C) system and etc are used in CNP1000 design for meeting safety requirements according to Chinese practical situations. PSA methodology can optimize CNP1000 design so as to reduce core melt frequency. Through use of increasing thermal margin of core the investment risk of plant owners can be reduced and it makes possible to offer more flexibility for operation and development of nuclear power plant. The paper will introduce the design features of CNP1000 and improvement measures compared with Daya Bay NPP. The international cooperation proposal and the ways of reducing construction and operation costs of CNP1000 are also discussed in the paper.

1. INTRODUCTION ABOUT CNP1000

China needs nuclear power for continuous development of its national economy. Now, about 6 units of nuclear power plant are planned to construct in near future at three sites of China. But many useful lessons from past constructions of nuclear power plants in China should be drawn by Chinese nuclear power industry. Nowadays there are three types of water reactors to be built in China. Following the construction and the operation of 11 current nuclear power units China still is not able mostly to achieve self-reliance design and equipment manufacture localization for more than 1000MWe NPP. So self-reliance design and equipment manufacture localization is nuclear power policy of China for near future NPP construction.

According to Chinese practical situations, such as manufacturing capacity of main equipment, self-reliance design ability, experience of current NPPs and so on, the safety improvement for CNP1000 will be considered in the following several areas from design point of view.

(a) Low power density reactor is employed to get large thermal safety margin for fuel assembly;
(b) Increase reliability and capacity of safety systems so that the safety functions of which should be improved;

(c) Large water inventory of primary coolant system is useful to mitigate the consequence of small LOCA and improve pressure stability during plant operation as well;
(d) Large containment with enough ventilation capacity is needed for withstanding influence upon containment integrity induced by loss of coolant accident and main steam line rupture accident;
(e) Digital Instrumentation and Control (I&C) system is utilized to increase reliability and NPP safety;
(f) In the design it is necessary to use large steam generator (SG), specially large secondary side volume of SG.

On the other hand, in the design and construction of CNP1000 the following 6 main measures will be employed to reduce cost of construction and operation of NPP.

(a) Half speed turbine/generator is used to increase efficiency of nuclear power plant compared with Daya Bay and QS-II NPPs;
(b) Optimized management and modularization designs of some systems are employed to short construction period of NPP;
(c) Unique type of reactor, standardization and batch productions are very important for next Chinese NPP constructions;
(d) Self-reliance design, equipment localization is one important way to reduce construction cost of nuclear power plant in the case of Chinese current conditions;
(e) High availability factor of NPP is designed for reduction of generation cost;
(f) Design lifetime of CNP1000 will prolongs to 60 years from 40 years compared with Daya Bay NPP.

Chinese government has already determined the technology way for nuclear power plant construction. 1000MWe pressurized water reactor with 300MWe per loop is the best choice of nuclear power plant based on the current situation of Chinese nuclear power industry. The proven technology should be fully utilized in CNP1000 design.

The main design targets of CNP1000 are listed in Table I. All of main design targets in Table I are accepted to improve safety features and economic benefits of CNP1000. The sorter construction period, the longer plant operation life time and the larger availability factor can reduce the construction and operation costs. The smaller reactor core melt frequency and the larger thermal safety margin of core could increase safety of nuclear power plant.
Considering the requirements of design targets, the main parameters of reactor and primary coolant system for CNP1000 are proposed and given in Table II. In order to make comparisons, Table II also gives the main parameters of reactor and primary coolant system for Daya Bay and Qin Shan II (QS-II) NPPs. The operation pressure of reactor and average temperature of primary coolant are the same for CNP1000, Daya Bay and QS-II respectively (15.5MPa and 310℃). The large difference is the size of the core. 177 fuel assembly core is utilized for CNP1000 design, and 157 fuel assembly core for Daya Bay NPP design.

TABLE I. MAIN DESIGN TARGETS OF CNP1000

Parameter	Unit	Value
Electric power output	MWe	Around 1000
Plant design life time	year	60
Availability factor	%	≥87
Refueling period	month	18
Plant construction period	month	≤66
Reactor core melt frequency	1/r • y	≤10^{-5}
Radioactive material release frequency	1/r • y	≤10^{-6}
Thermal safety margin of reactor core	%	≥15
Construction cost	$ /kW	<1500

TABLE II. MAIN PARAMETERS OF REACTOR AND PRIMARY COOLANT SYSTEM

Parameter	Unit	CNP1000	Daya Bay	QS-Ⅱ
Rated power of reactor	MWt	2895	2895	1930
Type of fuel assembly		AFA-3G	AFA-2G	AFA-2G
Fuel assembly number		177	157	121
Average linear power density of fuel rod	W/cm	165	186	160.9
Operation pressure of reactor	MPa	15.5	15.5	15.5
Best estimated flow rate	m^3/h	3×25000	3×23790	2×24290
Inlet/outlet temperature of reactor	℃	293.9/326.1	293/327	293.4/326.6
Total volume of pressurizer	m^3	45	40	36
Main steam pressure upstream of SG restrictor	MPa	6.79	6.76	6.71
Main steam flow rate	t/h	3×1959.3	3×1938	2×1951
Feed water temperature	℃	230	226	230

2. MAJOR IMPROVEMENTS OF CNP1000 DESIGN COMPARED WITH DAYA BAY NPP

2.1. Larger reactor core

Up to now, Daya Bay NPP uses 157 AFA2G fuel assembly core and 12 month cycle for fuel management. If Daya Bay NPP will perform 18 month fuel cycle and low neutron leakage, Departure from Nucleate Boiling Ratio (DNBR) can not meet the requirement of 15% thermal safety margin as shown in Table III. As known from the analyses of such two typical accidents as loss of flow rate and rod drop for AFA2G fuel assembly core, the reactor of Daya Bay NPP only has 5.3% thermal margin under 12 month cycle condition, but the departure from nucleate boiling will occur under 18 month cycle condition. It means the requirement of licensing and basic safety limit can not be met for Daya Bay AFA2G fuel assembly core under 18 months cycle condition. However, although the DNBR margin of the reactor can meet licensing and basic safety limit requirements but still is not able to achieve 15% if 157 AFA3G fuel assembly core performs 18 months cycle. That is why more fuel assemblies are needed in the core for CNP1000 compared with Daya Bay NPP.

TABLE III. CORE THERMAL SAFETY MARGIN ANALYSIS FOR DAYA BAY NPP

Parameters		12 months refueling AFA2G $F_{\triangle H}$=1.55	18 months refueling AFA2G $F_{\triangle H}$=1.65	18 months refueling AFA3G $F_{\triangle H}$=1.65
$DNBR^{min}$, nominal power		2.083	1.854	2.06
Loss of flow rate	$DNBR^{min}$	1.405	1.198	1.413
	DNBR margin	14.2%	-7.8%	8.7%
Rod drop accident	$DNBR^{min}$	1.421	1.236	1.458
	DNBR margin	5.3%	-9.1%	4.9%

In order to increase the safety margin based on Daya Bay NPP, the following two methods have been considered:

(a) Reducing thermal power of reactor core;
(b) Adding fuel assemblies in the core.

According to Chinese practical situation and future development of PWR NPP, the second method is chosen to increase safety of CNP1000. In this way the linear power density of core reduces to 165W/cm for CNP1000 from 186W/cm for Daya Bay NPP. 177 fuel assembly (AFA3G or Performance+) core is used in the CNP1000 design so as to increase safety of reactor and meet the design target requirement of 15% thermal margin.

2.2. Larger Reactor Pressure Vessel (RPV)

The reactor pressure vessel of CNP1000 becomes larger because of more fuel assemblies in the core compared with Daya Bay NPP as shown in Figure 1. The inner diameters of RPV are 3989mm for Daya Bay NPP and 4340mm for CNP1000 respectively. Large reactor pressure vessel not only adds water inventory above the core but also reduces neutron fluence on the wall of RPV. Therefore, CNP1000 has more water inventory capacity to mitigate consequences of some accidents especially such as small LOCA and meets the design target requirement of 60 year plant life time as well.

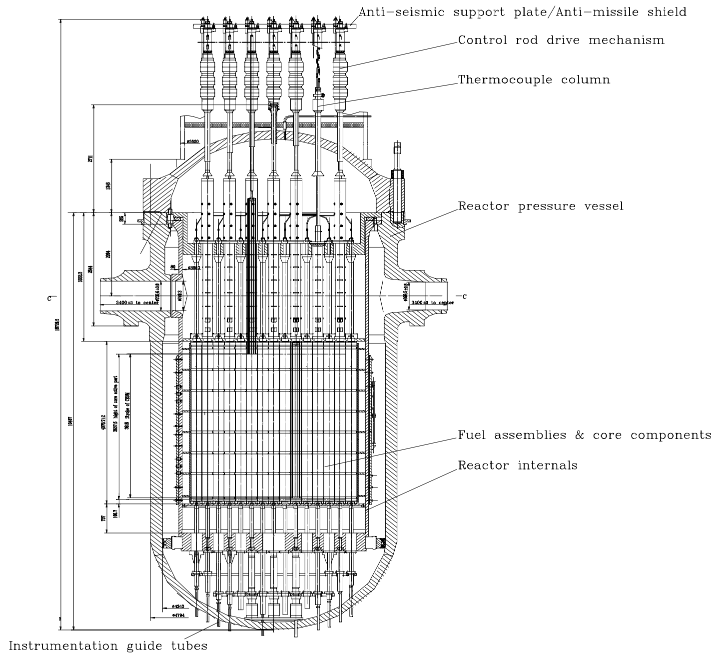

FIG. 1. CNP 1000 reactor structure.

Table IV gives calculation results of fast neutron fluence on the wall of RPV during the certain life time of nuclear power plant. It is seen from preliminary calculation that the fast neutron fluence on the inner wall of CNP1000 RPV is 2.9×10^{19} n/cm^2 during operation period of 60 years. According to Chinese current smelting technology of RPV material and experiment data, 4.0×10^{19} n/cm^2 is determined as the fast neutron fluence limit. As shown from the calculation, the fast neutron fluence on the wall of RPV is quite smaller for CNP1000 (Option I) than those for Option II and Option III.

TABLE IV. COMPARISONS OF FAST NEUTRON FLUENCE AND LIFE TIME FOR REACTOR PRESSURE VESSEL (RPV)

Parameters	Unit	Option I	Option II	Option III
Reactor power	MW	2895	2775	2895
Number of fuel assemblies in the core		177	157	157
Type of fuel assembly		AFA3G	AFA3G	AFA3G
Inner diameter of RPV	mm	4340	3989	3989
Design life time	year	60	50	48
Fast neutron fluence limit	n/cm^2	4.0×10^{19}	5.45×10^{19}	5.45×10^{19}
Fast neutron fluence on inner wall of RPV	n/cm^2	2.9×10^{19}	5.42×10^{19}	5.43×10^{19}

2.3. Integrated structure on reactor top

The structure on the reactor top including control rod drive mechanism (CRDM), ventilation, support and preventing missile shield is welded into an integrated assembly as shown in Figure 2. This structure has many advantages. A steel plate is used as both anti-seismic support and preventing missile shield, which can simplify structure. A barrel assembly acts as the support of reactor top so that reliability of the whole structure will be increased. The integrated structure on reactor top is useful to decrease the occupied site and space and also sort outage time as well. On the other hand, the integrated structure of CRDM seal shell assembly and RPV tube holder is able to make the pressure boundary of primary coolant system stronger and reduce the probability of loss of coolant accident.

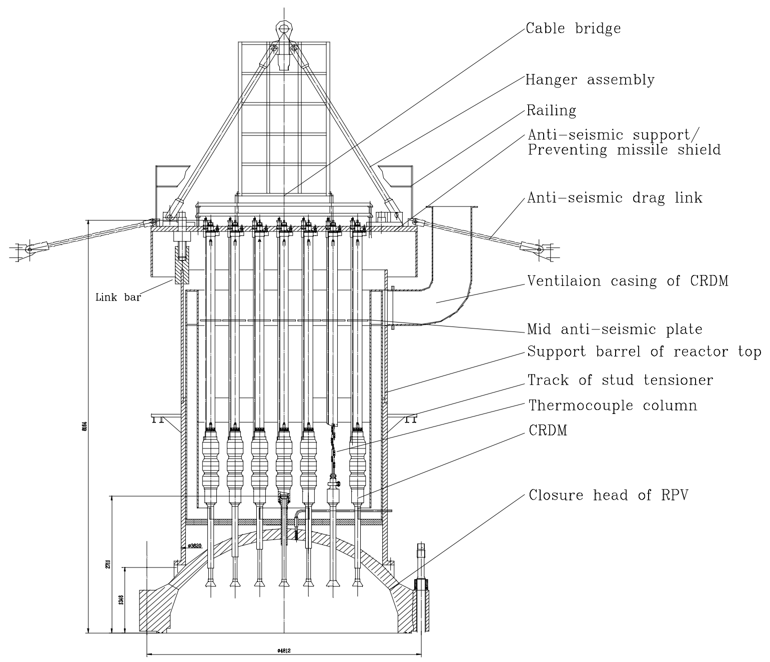

FIG. 2. Integrated structure on reactor top for CNP 1000.

2.4. Improved fuel management strategy

The following requirements and methods are considered in the nuclear design of CNP1000.

(a) The cycle length of 18 month refueling should be more than 470EFPD;
(b) 1/3 refueling, IN-OUT shifting and low neutron leakage is used in the fuel management design;
(c) U-235 enrichment is not allowed to be larger than 4.5%;
(d) Gd2O3 burnable poison is designed to compensate excess reactivity, to flatten power distribution and to keep enough negative temperature coefficient of reactivity;
(e) The enthalpy rise hot channel factor F△H and heat flux hot channel factor Fq are equal to or smaller than 1.65 and 2.45 respectively for CNP1000 (1.55 and 2.25 for Daya Bay NPP) according to preliminary analysis.

Table V gives fuel assembly number of each cycle refueling and U-235 enrichment of new fuel assemblies. Table VI gives comparisons of reload characteristics for AFA3G fuel assembly cores.

It is seen from analyses of nuclear design and fuel management that 18 month equilibrium refueling will be reached following the sixth cycle. 60,60 and 56 new fuel assemblies with 4.45% U-235 enrichment are respectively reloaded the core every cycle for 1/3 refueling. The cycle length is about 475EFPD. The reactivity temperature coefficient of moderator always keeps negative and shut down margin is more than 2000pcm. F△H and Fq meet design requirement.

TABLE V. LOAD NUMBER OF FUEL ASSEMBLY

Range	Enrichment %	Number of core cycle					
		1	2	3	4	5	6
1	1.8	57					
2	2.4	60	57				
3	3.1	60	60	57			
4	3.7		60	60	57		
5	4.45			60	60	57	
6	4.45				60	60	57
7	4.45					60	60
8	4.45						60

TABLE VI. COMPARISONS OF RELOAD CHARACTERISTICS FOR AFA3G FUEL ASSEMBLY CORE

Parameters	Unit	Option I	157 assembly core option		
Reactor power	MWt	2895	2895		
Number of fuel assemblies in the core		177	157		
U-235 enrichment	%	4.45	4.45	4.70	4.95
Number of reload fuel assemblies		60/60/56	68/72	64	52
Average assembly discharge burnup	MWd/tU	49600	43600	48000	57000
Maximum assembly burnup	MWd/tU	54600	49300	53500	64000

The design limits of average assembly discharge burnup and maximum assembly burnup are 52000MWd/tU and 57000MWd/tU respectively for AFA3G. The calculation data in Table VI show the average assembly burnup of CNP1000 is near 50000MWd/tU which is, generally speaking, the optimization operation discharge burnup. 177 AFA3G fuel assembly core is able fully to utilize the benefit of AFA3G and reasonable reload assembly number so as to get most economic efficiency.

2.5. Improved safety systems

The high pressure safety injection pumps are used as charge pumps of Chemical and Volume Control System (CVCS) while the low pressure safety injection pumps are separated from the pumps of Residual Heat Removal System (RHRS) for Daya Bay and QS-II NPPs. It is found through use of preliminary analysis of Probabilistic Safety Assessment (PSA) methodology that the reliability of safety injection system can be increased if high pressure safety injection pumps are not used as charge pumps of CVCS. So high pressure safety injection pumps are independently set up separating from charge pumps of CVCS and low pressure safety injection pumps are also used as those of RHRS in CNP1000 design as shown in Figure 3.
The preliminary analysis about main steam line rupture accident represents it possible to reduce boron concentration of water in the boron injection tank from 21000ppm to 7000ppm. This measure can not only simplify the system but also benefit the maintenance
In order to assure reliability of Auxiliary Feed Water System (AFWS) of CNP1000, two turbine driven pumps are installed, the capacity of each which is 100% of rated flow rate to supply to steam generator. Consequently, there are two independent and redundant subsystems. Figure 4 is schematic diagram of AFWS.

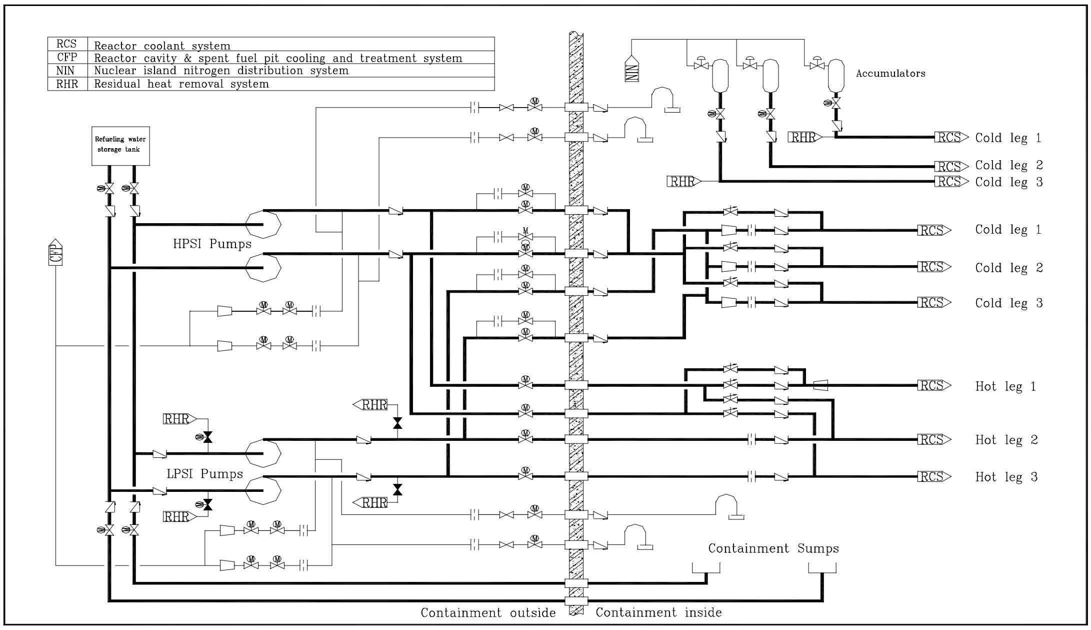

FIG. 3. Safety injection system.

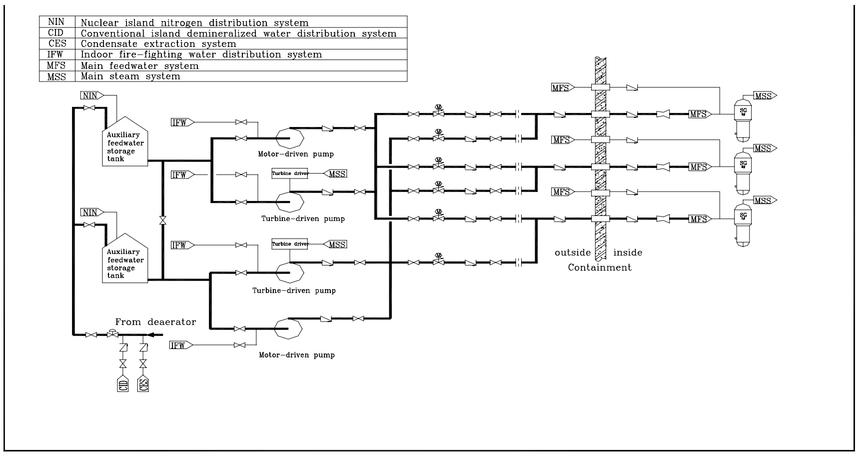

FIG. 4. Auxiliary feedwater system.

2.6. Digital Instrumentation and Control (I&C) System

Digital I&C of CNP1000 utilizes advanced technology of computer, communication, display, digital control and so on to improve safety and efficiency of control during plat operation. The technique process and operation monitoring of nuclear power plant, process control, power control, protection and safety monitoring during and following designated events are designed as an integrated architecture through use of computer technology, advanced human-machine interface and system engineering.

The centralized control, monitoring and the comprehensive management during operation of nuclear power plant are designed to increase reliability, maintainability and availability of CNP1000 I&C system and to improve safety and economy. The digital I&C system of CNP1000 is divided into three levels: management level, monitoring control level and input/output level, each of which is organically connected by fiber optic communication network. The digital I&C system contains the following three parts from consideration of structure point of view.

(a) Communication sub-system containing monitor bus which links sub-systems of importance and provides key information to the operator is a high speed, redundant communication network. The data exchange and transmission are performed through fiber optic cables with high noise immunity. The fiber optic cables can not couple electromagnetic interference or radio frequency interference noise into the system;

(b) Human-machine interface includes operation and control centers, data display and monitoring in the main control room. The graphics are supported by a set of microprocessor-based graphics workstations that take input data from the monitor bus. The distributed computer system delivers data over the monitor bus to other users. The control procedures implemented by operators to mitigate the consequences following beyond basic design accidents will be based on plant conditions such as reactor reactivity, water inventory and temperature of primary system, containment pressure and so on through use of computer systems in the main control room;

(c) Protection, control and monitoring sub-systems feed real-time data into the monitor bus for use by the main control room and computer system. Protection and safety monitoring sub-system is the aggregate of electrical and mechanical equipment which senses operation conditions and generates the signals to actuate reactor trip and engineered safety features, and which also provides the equipment necessary to monitor plant safety-related functions during and following accidents. Reactor control instrumentation sub-system mainly perform the control functions such as reactor power, rod position, water level and pressure of pressurizer, feed-water, steam dump and rapid power reduction.

3. CORE SAFETY MARGIN ANALYSIS

Under the guidance of CNP1000 design targets the following situations are analyzed to obtain an optimized reactor core which has more 15% thermal safety margin:

(a) Number of fuel assemblies: 177 and 157;
(b) Types of fuel assemblies: AFA3G and Performance+;
(c) Reactor power: 2895MWt and 2775MWt;
(d) Average coolant temperature: 310℃ and 307℃

Three main option analyses are listed in Table VII. The enthalpy rise hot channel factor F△H equals 1.65 in all of calculation analyses. In 12 month refueling cycle analysis the DNBR limit is 1.22 for full flow rate, 1.23 for loss of flow rate and 1.35 for rod drop accident calculated with statistical method respectively. In the 18 month refueling cycle analysis the DNBR limit value is 1.26 for full flow rate, 1.30 for loss of flow rate and 1.39 for rod drop accident calculated with statistical method respectively.

Tables VIII and IX show the calculation results of CNP1000 core DNBR margin corresponding to different conditions following accidents such as loss of flow rate and rod drop.

It is known from the calculation results provided by Table VIII that if reactor core utilizes 157 AFA3G fuel assemblies and average temperature of primary coolant still keeps 310℃ the DNBR margin is only able to achieve 13.2% following rod drop accident although reactor power reduces to 2775MWt from 2895MWt. Under the condition of lower reactor power (2775MWt), the DNBR margin can achieve 17.1% following rod drop accident and the design target of 15% safety margin can be met if average temperature of primary coolant reduces to 307℃ from 310℃. On the other hand, although lower reactor power and average temperature of primary coolant is designed to obtain larger thermal safety margin of the core, the electric power output of plant may decrease and the economic benefit could be lost very much. It stands to reason that 177 fuel assembly core is utilized in CNP1000 design. The reactor power and average temperature of primary coolant do not need to be reduced and more 15% DNBR margin can be achieved for 177 fuel assembly core as shown from Table VIII. Performance+ fuel assembly core has more DNBR margin than AFA3G core under the same reactor rated power. It is because the critical heat flux is larger for performance+ fuel assembly than for AFA3G.

TABLE VII. CALCULATION SCHEMES FOR CORE SAFETY MARGIN ANALYSIS

Parameters	Unit	Option I	Option II	Option III
Reactor power	MWt	2895	2775	2895
Number of fuel assemblies		177	157	157

TABLE VIII. COMPARISONS OF DNBR MARGIN
(AFA3G Fuel assembly core)

Parameters		Unit	Option I		Option II		Option III	
Reactor power		MWt	2895		2775		2895	
Number of fuel assemblies			177		157		157	
Average coolant temperature		℃	310	307	310	307	310	307
$DNBR^{min}$, nominal condition			2.32	2.40	2.2	2.27	2.06	2.11
Loss of flow rate	$DNBR^{min}$		1.699	1.75	1.572	1.627	1.413	1.52
	DNBR margin	%	30.7	34.6	20.9	25.1	8.7	16.9
Rod drop accident	$DNBR^{min}$		1.656	1.73	1.573	1.628	1.458	1.55
	DNBR margin	%	19.1	24.5	13.2	17.1	4.9	11.5

TABLE IX. COMPARISONS OF DNBR MARGIN
(Performance+ fuel assembly core)

Parameters		Unit	Option I		Option II		Option III	
Reactor power		MWt	2895		2775		2895	
Number of fuel assemblies			177		157		157	
Average coolant temperature		℃	310	307	310	307	310	307
$DNBR^{min}$, nominal condition			2.42	2.52	2.30	2.35	2.17	2.23
Loss of flow rate	$DNBR^{min}$		1.64	1.72	1.56	1.64	1.40	1.51
	DNBR margin	%	26.1	32.3	20.0	26.1	7.7	16.2
Rod drop accident	$DNBR^{min}$		1.75	1.83	1.66	1.73	1.56	1.64
	DNBR margin	%	25.8	31.7	19.4	24.4	12.2	17.9

TABLE X. COMPARISONS OF LINEAR POWER DENSITY

Parameters	Unit	Option I	Option II	Option III
Reactor power	MWt	2895	2775	2895
Number of fuel assemblies		177	157	157
Average linear power density	W/cm	165	178.3	186
Peak linear power density, 118% of nominal power	W/cm	482.9	526	548.7
Linear power density margin		22.1%	15.2%	11.5%

TABLE XI. COMPARISONS OF CENTER TEMPERATURE OF FUEL ROD

Parameters	Unit	Option I	Option II	Option III
Reactor power	MWt	2895	2775	2895
Number of fuel assemblies		177	157	157
Core peak power factor		2.48	2.5	2.5
Maximum center temperature of fuel rod under nominal condition	℃	1752	1908	1999
Maximum center temperature of fuel rod under 118% rated power condition	℃	1960	2133	2219
Temperature margin of fuel rod		24.3%	17.60%	14.3%

Besides DNBR margin analyses and comparisons, the linear power density and center temperature of CNP1000 core fuel rod have been also analyzed. Table X and XI give the calculation results and comparisons among options.

The linear power density margin of fuel rod is defined as:

(linear power density limit - peak linear power density) / linear power density limit

In CNP1000 design, the linear power density limit is 620 W/cm. The core with 177 fuel assemblies has more linear power density margin.

The temperature margin of fuel rod is defined as:

(Fuel temperature limit – maximum temperature) / fuel temperature limit

In CNP1000 design, the fuel temperature limit is 2590℃ when average assembly discharge burnup achieves 50000MWd/tU.

4. CALCULATION ANALYSIS OF CORE MELT FREQUENCY

Probabilistic Safety Assessment (PSA) methodology is employed to guide improvement of safety systems. 12 failure trees were established and 75 event sequences were calculated during assessment. In which 45 event sequences are able to cause core melt. It has been found from calculation analysis that:

(a) The frequency of core melt induced by two accidents of loss of offsite power and loss of component cooling water will reduce about 23% through using two turbine driven pumps in the auxiliary feed water system establishing two independent trains and subsystems;

(b) The frequency of core melt induced by loss of offsite power accident will reduce about one order (10 times) through adding the fifth diesel generator.

Table XII gives the calculation results of core melt frequency for CNP1000. The preliminary conclusions of core melt frequency analysis for CNP1000 can be obtained as the following two items.

TABLE XII. CALCULATION RESULTS OF CORE MELT FREQUENCY FOR CNP1000

Initial event	Frequency of initial event (1/r • y)	Core melt frequency (1/r • y)	Percentage (%)
Large LOCA	1×10^{-4}	7.99×10^{-8}	3.3
Mediate LOCA	3×10^{-4}	3.87×10^{-7}	16.1
Small LOCA	1.7×10^{-3}	1.17×10^{-6}	48.7
Extremely small LOCA	3×10^{-4}	4.11×10^{-7}	17.1
SGTR	6.5×10^{-3}	5.65×10^{-8}	2.4
Loss of offsite power	5×10^{-2}	6.16×10^{-9}	<1.0
Loss of component cooling water	7.49×10^{-5}	2.82×10^{-7}	11.7
ATWS	1.136×10^{-5}	8.53×10^{-9}	<1.0
Core melt frequency		2.40×10^{-6}	100

(a) 8 initial accidents and 75 event sequences were calculated in the above mentioned analyses without considerations of external events, operator failure, common failure and maintenance influence. So after consideration of those effects it seems possible that CNP1000 core melt frequency is in the range of **5×10^{-6}/r.y~8 $\times10^{-6}$/r.y**;

(b) CNP1000 reactor core has enough thermal safety margin, larger water inventory of primary system, improved engineered safety systems and digital I&C. Therefore, core melt frequency reduces and is less than 10^{-5} /r.y so that CNP1000 design meets target requirement.

5. ENGINEERING DEMONSTRATION EXPERIMENTS FOR CNP1000

The design and construction of Daya Bay and QS-II NPPs are for the reference of CNP1000 design. 177 fuel assemblies make up CNP1000 reactor core which adds 20 fuel assemblies compared with Daya Bay NPP reactor so that the stricture of CNP1000 core and RPV has been some changed. This change is reasonable and feasible according to PWR experiments and operations. Under the same other conditions, the larger core may be useful to flow distribution in the core and reduces flow-induced vibration of reactor internals. Nevertheless, at least the following two engineering experiments have already been arranged to demonstrate the feasibility of such change.

(a) Hydraulic simulation experiment, and
(b) Flow induced vibration experiment.

Furthermore, those two experiments will provide very useful and important data for the design. It is also possible to improve the structure of CNP1000 reactor internals after experiments. Hydraulic simulation experiment and flow induced vibration experiment will be finished by June and December, 2001 respectively so as to ensure the success of CNP1000 core change design.

6. INTERNATIONAL COOPERATION ABOUT CNP1000

During design and construction of CNP1000, it is necessary to cooperate between China and foreign countries. The range of international cooperation depends on the technology that will be utilized in CNP1000 design and major equipment manufacture. For example, if performance+ fuel assembly core is finally decided and used in CNP1000 NPP the computer codes and analysis methodology about nuclear design, fuel management, thermal hydraulic calculation and analyses of some accidents, product line and associated technology for performance+ fuel assembly manufacture should be transferred from foreign company to China.

Another example of international cooperation is about digital instrumentation and control system of CNP1000. Technology of digital I&C for nuclear power plant develops very rapidly. Chinese engineers and equipment suppliers still do not have enough experience to design and manufacture the advanced and reliable digital instrumentation and control system for CNP1000. So co-design, for instance, is in the consideration. The general technical responsibility of digital I&C design and technology transfer are borne by foreign company for first two units while Chinese engineers overall attend the design activities as much as possible. After then, the general technical responsibility will be borne by Chinese side from second two units while foreign company gives the technical support.

The design methodology and manufacture technology for some major components of CNP1000 is needed to transfer from foreign company so as to achieve equipment localization through constructions of 2 or 4 NPP units. The following is possible areas to make international cooperation:

(a) Primary coolant circulation pumps, specially, the technology of design and manufacture for shaft seal equipment and associated analysis software need international cooperation;
(b) Steam generator, specially, the technology of design and manufacture for moisture-separating equipment and hydraulic analysis codes need to transfer from foreign company;
(c) The design and manufacture technology for half speed turbine/generator;
(d) Digital I&C design and development.

Available cooperation has two models due to current Chinese practical situation.

(a) Co-Design Model
The co-design model enhances localization, and is desirable to make a stronger and larger Chinese leadership role based on the current Chinese nuclear technology and government localization policy. The co-design model can be applied to CNP1000 design, with minimal risk to the owner and a high degree of engineering and manufacturing localization. Foreign company will bear overall technical responsibility for CNP1000 design and Chinese engineers will fully join the CNP1000 design activities with foreign designers in the co-design model.
(b) Chinese Indigenous Design with Foreign Consultation
In this model, foreign company will provide technical consultation in the system and equipment design and related procurement to Chinese design institutes on CNP1000. If necessary, foreign company will provide the components in accordance with the Chinese specification and technical support to the owner for startup testing.

Cost reduction and safety design features of new nuclear power plants in India

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Abstract. Indian Nuclear Power Programme is designed to exploit limited reserves of uranium and extensive resource of thorium. Pressurised heavy water reactors are found most suitable and form the main stay of the first stage of the programme. Thorium utilisation is achieved in the second & third stages. Today India has total installed capacity of 2720 MWe of PHWRs which are operating with high plant load factors of over 80%. Rich experience of construction and operation of over 150 reactor years is being utilised in effecting cost reduction and safety improvements. Standardisation and reduction in gestation period by preproject activities, advance procurement and work packages of engineer, procure, construct and commission are some of the techniques being adopted for cost reduction in the new projects. But the cost of safety is rising. Design basis event of double ended guillotine rupture of primary pressure boundary needs a relook based on current knowledge of material behaviour. This event appears improbable. Similarly some of the safety related systems like closed loop cooling water operating at low temperature and pressure, and low usage factors may be designed as per standard codes without invoking special nuclear requirements. The paper will address these issues and highlight the possible areas for cost reduction both in operating and safety systems. Modern construction and project management techniques are being employed. Gestation period of 5 years and cost of less than US $1400 per KWe are the present targets. In Indian environment nuclear power is found to be competitive with thermal power plants at distances of about 800 Kms from the coal mines.

1. INTRODUCTION

Indian Nuclear Power Programme is based on a 3-stage strategy of exploiting limited reserves of uranium but large reserves of thorium. The first stage employs Pressurised Heavy Water Reactor (PHWR) as its mainstay for producing electricity in an economic and safe manner. First PHWR of 220 MWe capacity was set up in India at Kota, Rajasthan in 1972, a collaborative venture with Canada. Since then 11 more reactors of similar capacity have been constructed and commissioned. Today India has total installed capacity of 2720 MWe providing much needed electricity to its people. The performance of these units has been improving as seen from the rising trend of average annual Plant Load Factor (PLF) of 60% in 1995 to over 80% in the year 2000. This has been possible by better grid management, training and advanced maintenance practices. Sharing of International experience through WANO & COG has made significant contributions in strengthening these areas.

During construction of these plants, India has gone through distinct phases of learning, indigenisation, consolidation and standardisation. Improvements in design and cost effective safety features have been the key thrust areas on engineering side. New and innovative construction techniques like slip forming, pumped concrete, top lowering of equipment in Reactor Building by using heavy duty cranes have been evolved to reduce the construction period and interest during construction. Serial or construction of cluster of units at a given site is seen to have tremendous benefits in speedy execution of the units once work on first unit starts. This was effectively brought home during the construction and commissioning of recent units at Rajasthan and Kaiga where lot of innovative initiatives were taken to drastically reduce erection & commissioning periods.

Currently, construction work on two more units of 220 MWe each is expected to start at Kaiga. These reactors follow standardized design and meet latest safety requirements. Significant improvements have taken place in the design safety features of our Nuclear Power Plants. From dousing system based on active components to passive suppression pool for limiting containment peak pressure; from slow acting moderator dumping in RAPS/MAPS to fast acting dual redundant and diverse reactor shutdown system in the subsequent units; from emergency core cooling system provided by moderator cooling system to light water Emergency Core Cooling System backed by commensurate support systems qualified for seismic and Loss of Coolant Accident environment; from single to double containment; enhanced emergency power supply are some of the striking examples of raising safety standard compliance as per international practice. These features and a few others related to core flux mapping, architecture, lay-out for constructibility and maintainability have been incorporated in the evolutionary design of higher sized units of nominal rating of 500 MWe each, two of which are under construction at Tarapur. The estimated cost, including cost of capital is about US $1400 per KWe installed. Further cost reductions are expected as intense efforts are being made to cut down the construction schedule by adopting newer project management techniques which have been successfully employed in the construction of thermal power plants. Outsourcing, EPC (Engineer, Procure & Construct) and modern construction & erection techniques are expected to limit the construction period to a little over 5 years from first pour of concrete. The Government is also encouraging Nuclear Power Programme as it has adopted a policy of judicious mix of Thermal, Hydel and Nuclear as bulk supplier of electricity. Certain taxation benefits have been provided to accelerate augmentation of nuclear capacity. In addition it is perceived to be clean, environment friendly, green source of power for fulfilling aims of the agreements reached internationally for climate control. Nuclear power is also economical and competitive when located about 800 Kms away from the coal-mines which are largely located in the eastern part of India.

The paper will describe Indian nuclear power programme with over 150 Reactor years of operating experience and highlight cost reduction and design safety features of new nuclear power plants.

2. INDIAN NUCLEAR POWER PROGRAMME

As already mentioned, Dr. Homi Bhabha, the founding father of atomic energy formulated a long term strategy for nuclear power programme for judicious utilization of our limited reserves of uranium and vast reserves of thorium. The strategy was based on development in three stages, linking the fuel cycle of Pressurised Heavy Water Reactor (PHWR) and Fast Breeder Reactor (FBR).

First Phase comprises series of Pressurised Heavy Water Reactors (PHWR) using natural Uranium as fuel and heavy water as moderator and coolant. These reactors apart from very efficient utilization of fuel also produce small amount of Plutonium, required for the next phase of the programme, based on fast breeder technology. Use of natural uranium fuel enabled setting up of fuel fabrication facilities, without the need of capital intensive uranium enrichment plant that would have been necessary for light water reactors. It is proposed to build 10,000 MWe of nuclear power plants of PHWR type to form a good base for the second stage. The known uranium reserves of 70,000 tonnes will sustain this programme for about 50 years.

Second phase is to utilize the plutonium generated in first phase, in fast breeder reactor (FBR) wherein thorium can be converted into fissile material U-233. Thus, in this phase, our large reserves of thorium amounting to over 350,000 tonnes start getting utilized. A prototype Fast Breeder Test Reactor (FBTR), of 40 MW (th) capacity built with French support, at Indira Gandhi Centre for Atomic Research (IGCAR) is providing valuable design and operating experience in Sodium coolant technology. Indigenous design of 500 MWe Prototype FBR is nearly completed at IGCAR and first such unit is expected to be operating by the turn of this decade. There are many challenges in the development of this advanced technology, which calls for strong commitment both in terms of material and human resource.

Third phase is to utilize Uranium 233 in either fast reactors or in Advanced Heavy Water Reactors (AHWRs) where it can be mixed with thorium to operate in self-sustaining mode. Design of an Advanced Heavy Water Reactor having state-of-the-art passive design safety features is currently in progress. It is being developed entirely as an indigenous effort.

Fig.1 depicts three stages of Indian NPP in a pictorial form.

Performance of Nuclear Power Plants (NPPs):

The generating performance of NPPs has seen progressive improvement since 1995-96. The Plant Load Factor of NPCIL stations for the last six years are given in the Table I below.

TABLE I. HIGHLIGHTS OF PERFORMANCE DURING LAST 6 YEARS

	1995-96	1996-97	1997-98	1998-99	1999-00	2000-01
Nuclear Power Capacity (MWe)	1540	1840	1840	1840	2060	2720
% of installed capacity	1.86	1.83	1.83	2.1	2.3	2.5
Capacity Factor (%)	60	67	71	75	80	82.5

During the current year 2000-2001, the PLF is 82.46% . Continuous rising trend has tremendously boosted the profitability, commercial viability and the morale of the people. This would have been even higher if accounting practices had permitted refixation of tarrif for the older plants that sell electricity at one third to one half of current tarrifs.

Even though % of installed capacity is 2.5%; nuclear power stations contribute over 4% of electrical energy needs. Figure-2 shows share of different sources of power. NPCIL has also been successful in refurbishment work and the coolant channel replacement and upgradation programme of RAPS-2 (200 MWe). This was successfully carried out based on indigenous efforts and the unit was brought back on line. TAPS-1&2 (2 x 160 MWe) units established in 1969 are operating well at high capacity factors and plant life extension programme is progressively being implemented.

Present Status of Nuclear Power Programme

Nuclear Power Capacity in Operation:

Present total nuclear power capacity is 2720 MWe as of end March 2001 with 14 units connected to the grid. The details are in Table II.

Ongoing Projects under Construction:

Presently the indigenously designed and developed 2 x 500 MWe PHWR units are under construction at Tarapur, which when commercially operational in the year 2006 and 2007 will take the installed capacity to 3720 MWe.

Projects in Pipeline

The proposal for the project financial sanction for setting up of the 2 x 1000 MWe VVERs at Kudankulam with Russian cooperation is expected to commence after the DPR is completed and evaluation of the techno-commercial offer and their approval by Government of India.

Construction work on the 500 MWe PFBR, which will be a prototype unit for the Fast Breeder Reactor Programme, is planned to start next year.

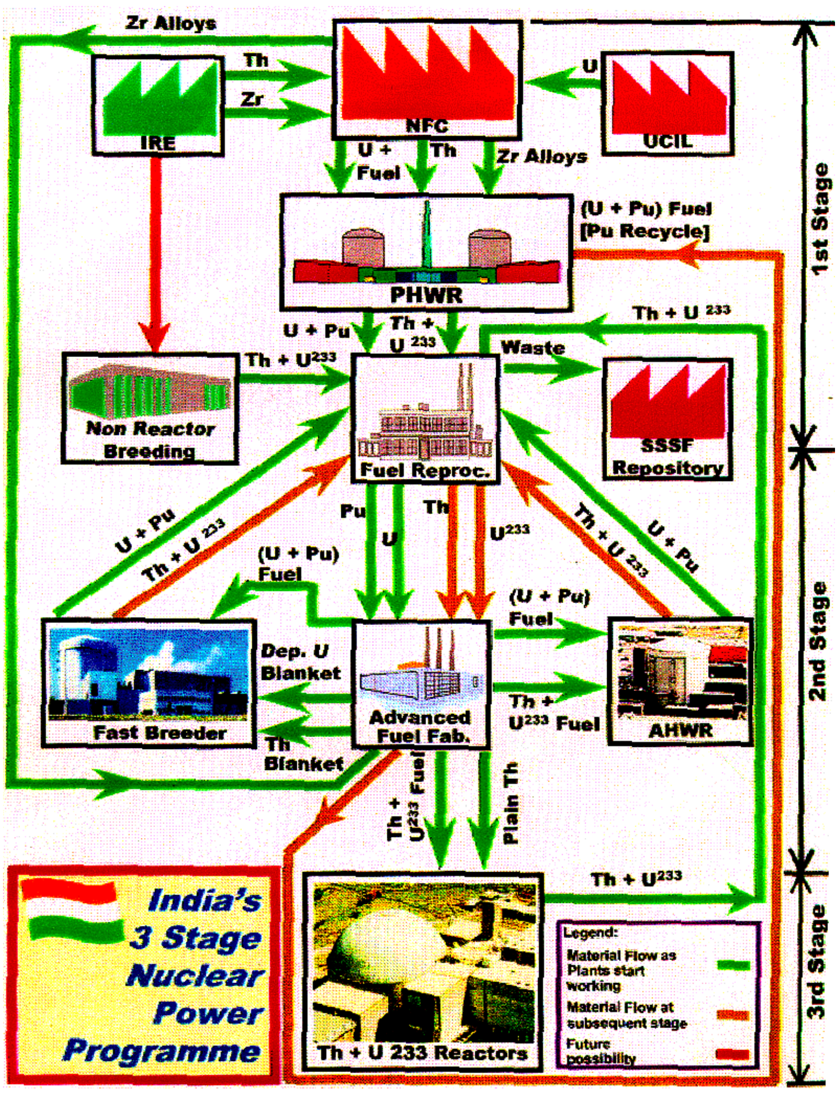

FIG. 1. India's 3 stage Nuclear Power Programme.

TABLE II. PLANTS UNDER OPERATION

PLANT	RERATED CAPACITY (MWe)	COMMERCIAL OPERATION SINCE
TAPS-1&2	2 X 160	October 28, 1969
RAPS-1	1 x 100*	December 16, 1973
RAPS-2	1 x 200	April 01, 1981
MAPS-1	1 x 170	January 27, 1984
MAPS-2	1 x 170	March 21, 1986
NAPS-1&2	2 x 220	January 01, 1991 & July 01, 1992
KAPS-1&2	2 x 220	May 06, 1993 & September 01, 1995
KAIGA-2	1 x 220	March 16, 2000
RAPS-3	1 x 220	June 01, 2000
KAIGA-1	1 x 220	November 16, 2000
RAPS-4	1 x 220	December 23, 2000

* Presently operating at 150 MWe with the clearance of regulatory authority

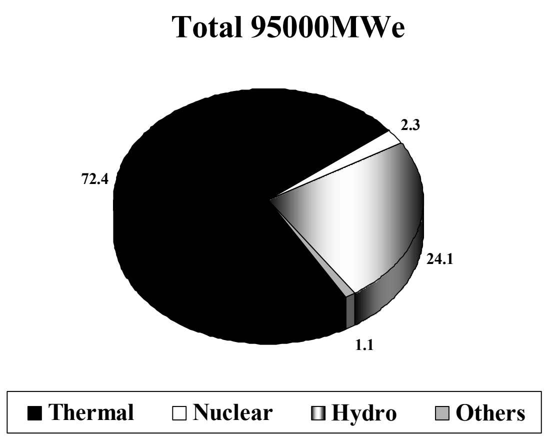

FIG. 2. Share of nuclear power in India – Year 2000.

New Nuclear Power Projects:
It is proposed to commence construction on the following projects:
RAPP-5 to 8 - 4 x 500 MWe PHWRs
Kaiga-5&6 - 2 x 220 MWe PHWRs

Nuclear Power Capacity Build-up to the end of coming decade is summarised in Table III:

TABLE III. NUCLEAR POWER CAPACITY BUILDUP

Details of Reactor Units	**Total Capacity**
Operating Reactors: 2 BWRs of 160 MWe each, 10 PHWRs (1x100, 1x200, 2 x 170, 8 x 220 MWe)	2,720
Reactor Under Construction: (2 PHWRs of 500 MWe each)	1,000
Reactor for which project activities have started: (2x220 + 2 x 1000 MWe)	2,440
Sub total	**6,160**
Reactors planned: (2x 220 + 4 x 500 MWe)	2,440
Reactors to be Planned: (6 x 500 MWe)	3,000
Sub total	**5,440**
Grand Total	**11,600**

3. COST CONSIDERATIONS

3.1. Cost reduction means and needs

In a power-starved country like India, no power is costlier than no power for meeting the needs of industrial development. But there is no denying to the fact that for long term survival in a market driven economy, nuclear industry must compete with other bulk producers, like hydel and thermal. Currently though, the natural gas based plant with overall plant efficiencies of 50 to 60%, are setting the bench marks for future development of nuclear energy. Because of the short supply of gas and its rising cost, CCGT (Combined Cycle Gas Turbine) plants have not made great inroads into Indian power sector. While cost of production is a basic ingredient for financial evaluation, it is seen that location of the plant from coal mines is a vital parameter in making investment decisions. A nuclear power plant at a distance of over 800-Km from source of coal is found competitive with thermal power plant. Further refinement of cost comparison with plant life of 60 years (30 years for thermal) and low operating nuclear fuel cost is carried out on discounted cash flow method. It provides clear basis for making decision.

Life extension of nuclear power plants as mentioned earlier, is of much interest to us as with a minimum investment and time period, the life of an existing plant is nearly doubled. Enmasse coolant channel replacement at RAPS-2 of 220 MWe along with safety upgrade and ageing management has given rich experience in plant life extension. Similar exercise is planned for subsequent plants at regular intervals because of limited life of Zircaloy-2 coolant tubes and problem in its supporting arrangement in nuclear environment. Zirconium-2.5-Niobium is the new alloy for coolant tubes having long life and enhanced mechanical and nuclear properties.

Safety upgrades and refurbishment include introduction of high pressure ECCS injection, simplification of dousing system from modulating to on-off duty, secondary/supplementary control centre, segregation of safety related cables into two independent groups to preclude common cause failure, installation of additional DG set for management of floods resulting from the failure of upstream dam, improving fire protection system etc. Ageing management focussed in replacement of heavy water heat exchangers, condenser tubes, building coolers and impellers of important pumps. Total refurbishment and coolant channel replacement was done at about 20% the cost of a new plant in about 30 months.

There are obvious advantages in exploiting an existing site for further construction of new plants. The time and finance requirement for developing a new site, rehabilitation of displaced persons, political compulsions and infrastructure, favour setting up of new plants at existing sites. The translation of this approach is being witnessed in planning of four more units at Kaiga (two already commissioned), six more units at Rajasthan (two more commissioned), augmentation at Kalpakkam, Narora and Kakrapar.

There is an urgent need to reduce the cost of construction of newer plants for the survival of nuclear industry in the immediate future. An optimal design with maximum application of local technology, catalogue engineering and configuration management are the fundamental basic inputs for capital cost reduction. The architecture and design must consider constructability, maintainability and inspectability aspects. Features for simultaneous civil and mechanical erection; routing and laydown areas for equipment maintenance; shielding for access in radioactive areas and maneuverability for Inservice Inspection (ISI) are incorporated in the design of current Indian PHWR. Large openings have been provided in the top of reactor dome for erection and possible future replacement of steam generators which may be required to be replaced once in life time based on current international and Indian experience with respect to Steam Generator tube behaviour.

Founding fathers of Indian nuclear power programme made the road map for development of this source of energy with focus on indigenisation and self-reliance. This has been the corner stone of our policy and has been responsible for creating large pool of trained manpower in nuclear science and technology. They are providing the required support not only to Indian programme but also assisting the developing countries in specific fields. However indigenisation and technology development slowed down initial growth. The trend in construction time span is graphically represented in Figure 3. The interest burden was obviously high. With the opening of the economy and increased competition, the capital cost had to be brought down emphasizing reduction in the gestation period.

Several steps in expediting design, procurement, construction and commissioning activities have been taken to achieve this. These are listed below:

Design:

- Freezing of bulk design before start of construction
- Reduction in number of welds by engineering custom built pipes and increased shop assemblies.
- Application of 3-D models for checking interferences.
- Catalogue engineering and use of commercially available equipment.
- Critical review of specified design tolerances and construction materials.

Procurement:

- Vendor prequalification and standardisation

- Adoption of work package approach for balance engineering, procurement, and erection and commissioning. The whole project is divided into about 50 packages. The basic advantage is that single point responsibility is fixed on the vendor.
- Advance procurement action is taken for long-delivery items such as End Shields, Calandria, Steam Generators, Reactor Coolant Pumps, Pressuriser, Heavy Water heat exchangers.

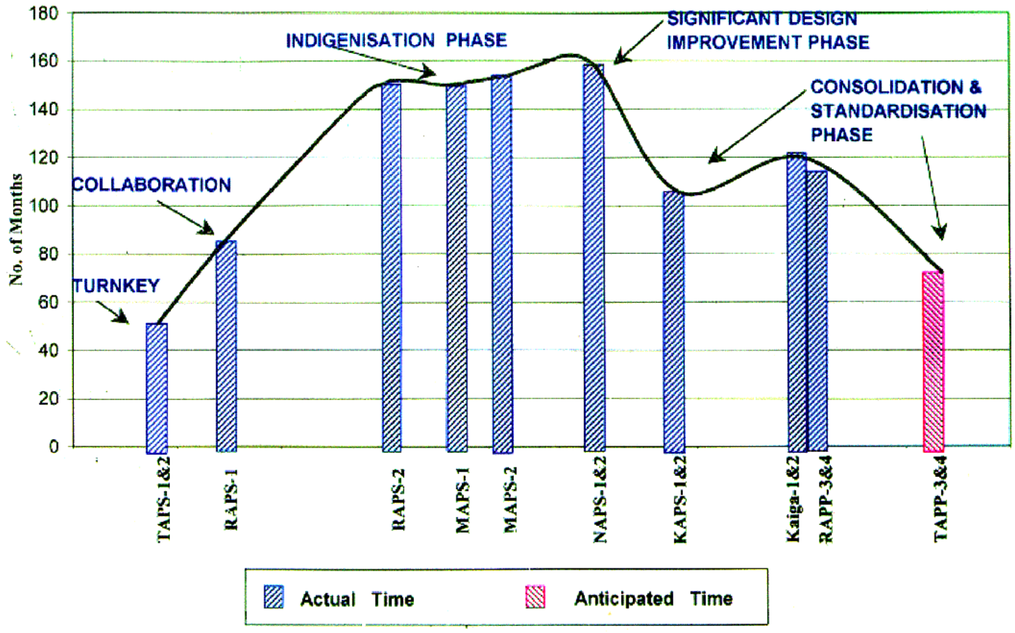

FIG. 3. Trend in construction time span in India nuclear power plants.

Construction:

- Pre-project activities like development of site infrastructure, excavation are carried out before first pour of concrete in the raft of reactor building.
- Increased mechanization (see Figure 4).
- Built up provision in design to facilitate equipment erection.
- Standardization
- Construction of multiple units at a site to save on site acquisition cost and take benefit of established infrastructure like construction water & electricity supply, access road, construction machinery and workshops.

3.2. Criticality to commercial operation

Once the plant is constructed and systems commissioned individually, a major milestone is to attain criticality after necessary clearance by the independent regulatory board.

In our earlier plants when we were in the learning phase we took as high as 20 months from criticality to synchronization. Time was spent in carrying out reactor physics experiments and generation of database for different configurations. Similarly, a cautious approach was followed in raising the power level. The unit was operated at various power levels and system performance evaluated before going to next stage. Over four months were spent to declare the unit commercial. In contrast RAPP-4, the most recent unit to be connected to the grid, only 14 days were spent to synchronise the station after first criticality. The reduction in period was largely due to precommissioning of services, cleanliness of systems and parallel working on steam systems.

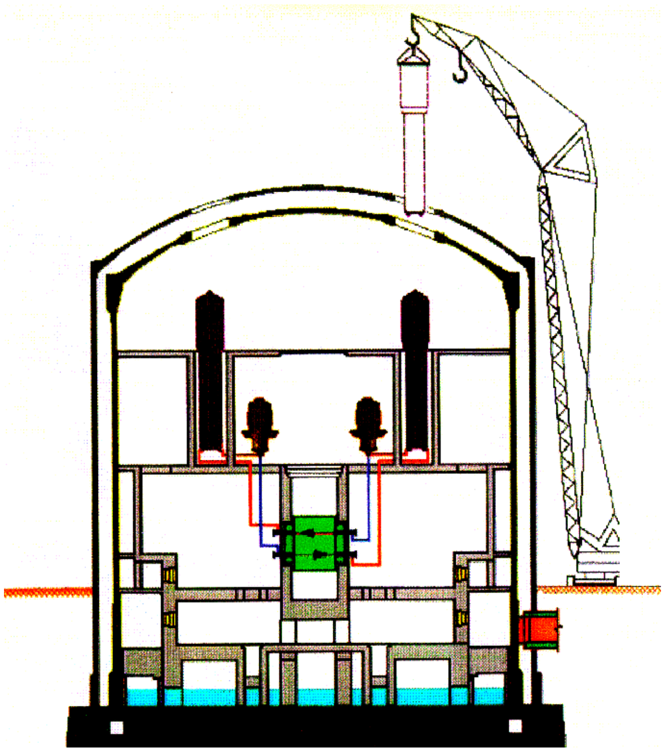

FIG. 4. PHWR double containment.

4. SAFETY OF NUCLEAR POWER PLANT

Some of the safety features of Indian PHWRs are :

Reactor shutdown systems:

- Two diverse, independent automatic fast acting shutdown systems for guaranteed shutdown viz: Mech. Shut off rods and Liquid Poison Injection
- Each system is fast enough to safely terminate worst reactivity transient (including voiding during LOCA)
- Failsafe design.

The evolution of reactor shutdown system has followed the following stages:

- Moderator dump was used in the earlier designs at RAPS & MAPS.
- From NAPS onwards two diverse fast-acting shutdown systems as described above were introduced.
- From Kakrapar onwards independent sensors and instrument channels for two systems were incorporated.
- However improvements were introduced at Kaiga/RAPP which are the latest 220 MWe Plants. On-line test facility was enhanced.

Core cooling provisions:

The Main Coolant Pumps are fitted with flywheels to maintain adequate core flow, following loss of power before thermosyphoning takes over. In case of a loss of coolant accident, high pressure all header injection followed by long term recirculation ensures no significant fuel failures. Heavy water in calandria has sufficient subcooling margin, to maintain coolant channel geometry, should the emergency injection for some reason fails.

Containment:

All PHWR plants, with the exception of RAPS and MAPS, follow a double containment philosophy. Also with the exception of RAPS which has a dousing for vapour suppression, subsequent reactors have suppression pool for energy management following LOCA and restricting containment peak pressure. The containment has also been provided with engineered safety features such as filtration and pump back, secondary containment clean up and purge system to enhance mitigating capabilities.

It is seen that significant changes and improvements have been incorporated in the design of safety and safety support systems. The systems which support the containment are designed as per safety class-II. All these are low pressure low temperature systems and do not see any cyclic loads. We feel that to cut down the cost these systems can be designed as per standards and codes for non-nuclear components. It should be emphasized here that the cost of a given component immediately shoots up as soon as it is categorized as a nuclear grade component.

5. CONCLUDING REMARKS

Indian nuclear industry with installed capacity of 2720 MWe and plans to augment this to over 11000 MWe in another decade has matured. Over the years it has gone through learning, indigenisation and consolidation phase with constant technological upgradation for enhanced safety. Upfront licensing, standardisation and planning of multiple units at a site with increased mechanization are some of the steps to cut down the gestation period. New plants are aimed to be constructed in about 5 years. Certain safety issues are also being addressed to make the plant safety simple and cost effective. Nuclear Power Corporation of India supported by sister concerns and Indian industry is taking giant strides and making relentless efforts in further improving the viability of nuclear power which is clean and safe.

Annex 14

The use of probabilistic safety analysis in design and operation — Lessons learned from Sizewell B

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Abstract. Probabilistic Safety Assessments (PSAs) have been used extensively in the design and licensing of Sizewell B. This paper outlines the role of PSA in the UK licensing process and describes how it has been applied to Sizewell B during both the pre-construction and pre-operational phases. From this experience a "Living PSA" has been formulated which continues be used to support operation. The application of PSA to Sizewell B has demonstrated that it is a powerful tool with potential for future use. Its strengths and limitations as a tool need to recognised by both users and regulators. It is not a fully mechanistic means of ensuring design safety, but is an important aid to decision making. It also has the potential to allow risk judgements to be taken in conjunction with commercial and environmental issues.

1. INTRODUCTION

The use of risk assessment in support of nuclear power plant safety has been common practice in the UK for some time. Indeed risk assessment is a fundamental feature of safety management in the UK (see for example Ref. [1]). Legislation places the responsibility for safety upon the owner and operator of any enterprise. They are legally required to reduce the risk posed to both the public and employees to a level that is "as low as is reasonably practicable" (the ALARP principle). "Reasonably practicable" is not defined in legislation, but has been established in the courts as a result of cases brought under health and safety law. Although the fundamental requirements are based on risk, quantitative risk assessment is not always required. Many risk assessments are qualitative and deterministic criteria have been developed as a means of assuring safety. However, such criteria are, in principle, surrogates for risk criteria.

Probabilistic Safety Assessment (PSA) is now a well-recognised and established technique, which has been put to a range of uses. It started as a technique to quantify the "residual risk", beyond the design basis (and was more commonly referred to as Probabilistic Risk Assessment). As such it was seen as being complementary to the deterministic design basis approach. The implicit assumption was that the deterministic approach leads to an acceptable risk within the design basis envelope so the only question was what fell outside this. The most famous early example of a comprehensive PRA was the Reactor Safety Study (Ref. [2]) - WASH1400. This provided a level 3 PSA for two "typical" LWRs. Although much of the early work was aimed at a full quantification of the risk to members of the public it was recognised that the intermediate results (core damage frequencies and containment failure frequency) were in themselves useful measures of system performance, though care had to be exercised since the sequences which dominate one measure may not always dominate others. WASH 1400 (Ref. [2]) had illustrated this demonstrating the importance of the interfacing systems LOCA to overall risk even though it was not a particularly important contributor to the core damage frequency.

Workers in the UK were involved in PSA work from the start and one of the earliest attempts to define an "acceptable" risk profile was the so-called "Farmer criterion" (Ref. [3]), which is still used in various forms today. Analysis of the Windscale fire in the UK lead to the development of codes to quantify accident consequences which has lead to tendency to use level 3 PSAs to a greater extent than is common elsewhere.

The use of probabilistic targets as part of design safety criteria has developed in the UK over a long period of time. Indeed there is a sense in which the public documentation of this has tended to lag behind actual practice. So that when the, then, CEGB and NII set down their Design Safety Criteria (DSCs) (Ref. [4]) and Safety Assessment Principles (SAPs) (Ref. [5]) in the late 1970s this marks, not the start of the use of such criteria but their formalisation. Probabilistic techniques and targets were used in the system design of the Advanced Gas Cooled Reactors (AGR) from the early 1970s. By the time the Sizewell 'B' project was started the use of such techniques was widespread having been extensively applied to the later AGRs. However even during this project the role of PSA has developed, and its use in the operational phase has continued, so it provides a useful case study.

2. UK LICENSING APPROACH FOR NEW PLANTS

To build and operate a nuclear power plant (NPP) in the UK requires a single licence, which covers construction, operation and decommissioning. The nuclear site licence is a standard one that requires arrangements to be in place, at all stages, to comply with the (36) licence conditions. "Hold points" during construction and commissioning, are defined and agreed, between the regulator and the licensee, which provide a formal means of control during these early stages. Controls through the lifetime of the plant are exercised by means of routine auditing and monitoring, restart consents following outages and periodic safety reviews, backed up by the ability to issue licensing instruments and directions at any time. The licensing approach is non-prescriptive but the NII's powers are wide-ranging and extensive.

To obtain a license to construct and operate a NPP, three submissions are generally required. These are:

i) A Preliminary Safety Report (PSR), which demonstrates the feasibility, in principle, of the design;
ii) A Pre-construction Safety Report (PCSR). This report provides the basis on which the Nuclear Site Licence is granted. As such the NII require it to contain sufficient detail for them to be confident that there will be no significant design changes required, for safety reasons in the design development phase. The level of detail required is therefore more equivalent to that required for final design certification in the US rather than that required for a construction licence;
iii) A Pre-Operational Safety Report (POSR). This provides the justification for the operation of the plant and builds on the PCSR. In the case of Sizewell B this had to submitted to NII 1 year before fuel load. (It should be noted that NII did not formally approve the POSR, they simply gave consent to load fuel – after a very thorough review of the safety case.) Following the completion of commissioning, the results of the commissioning and start-up tests are fed back into the POSR, which then becomes the Station Safety Report (SSR). For Sizewell B this is maintained as a living document and is updated to reflect modifications as they are implemented and is subjected to a wide-ranging "Periodic Safety Review" every 10 years.

In the case of Sizewell B a specific PSR was not produced. A wide ranging review, sponsored by the government and involving CEGB, NII and UK industry, was carried out to determine the future thermal reactor build. This Thermal Reactor Study, which led to selection of a PWR, was deemed to fulfil all the requirements of a PSR.

3. USE OF PSA IN THE DESIGN & CONSTRUCTION PHASES

During the Sizewell ‘B’ project PSA has been used in design, licensing and during the Public Inquiry process. During the project major PSA studies have been produced on three occasions:

i) a level 1 PSA for a range of internal initiators at power - during the design phase which forms part of the Pre-Construction Safety Report (PCSR);
ii) a level 3 PSA for a range of internal initiators at power, which formed part of the evidence, presented to the Public Inquiry;
iii) a comprehensive level 3 PSA for all initiators at all power levels (including shutdown) which formed part of the Pre-Operational Safety Report (POSR).

The Sizewell B licensing process and Public Inquiry led to the development both of the use of PSA and to refinement of the criteria against which such studies and the plant to which they relate are judged. The Sizewell ‘B’ Public Inquiry discussed both the DSCs and SAPs, and the underlying high-level criteria on which they were based, at length. Subsequently following the recommendation of the Public Inquiry Inspector the Health and Safety Executive (HSE) produced a discussion document (Ref. [6]) which set out to “formulate and publish guidelines on the tolerable levels of individual and social risk to workers and the public from nuclear power stations”. This in turn led to the publication by NII of revised Safety Assessment Principles for Nuclear Plants (Ref. [7]).

3.1. Pre-construction design phase

Both the CEGB DSCs and the NII SAPs were based on fundamental principles derived from the recommendations of the International Commission on Radiological Protection. These principles still form the basis of the current NII and British Energy safety principles. They embody the requirements both to satisfy the statutory dose limits and the ALARP principle (i.e. the risk must be reduced to a level which is as low as reasonably practicable (ALARP).

The Design Safety Criteria and Guidelines against which Sizewell ‘B’ was designed were formulated to ensure that these fundamental principles were satisfied. The underlying criterion on which the CEGB limits were based was that an increase in risk of fatality of 10^{6}/yr. was considered to be broadly acceptable. The probabilistic “criteria” were set as targets and were deliberately set at levels, which would be challenging to achieve to ensure that the design was ALARP. In general, where there is comparability, the DSC targets correspond to the Basic Safety Objective (BSO) of the new SAPs i.e. the bottom end of the ALARP region as defined by the Tolerability of Risk (TOR) paper.

Probabilistic targets were set both for small releases within the design basis and for “uncontrolled” releases. These were originally defined in terms of the lower limits of the Emergency Reference Level (ERL) for evacuation, since it was also the design intent that there should be no requirement for offsite evacuation for any design basis fault. In addition to the targets for small releases the DSC sets frequency targets for “uncontrolled releases” for both single accidents, which could give rise to a large uncontrolled release ($<10^{-7}$/yr), and for the sum of all such accidents ($<10^{-6}$/yr). For the purposes of the design and assessment a large uncontrolled release is taken to be a release that exceeds the whole body lower limit ERL (100mSv). Using this definition, the DSC levels can be directly compared with those of the SAPs (Ref. [7]) as is shown in Figure 1.

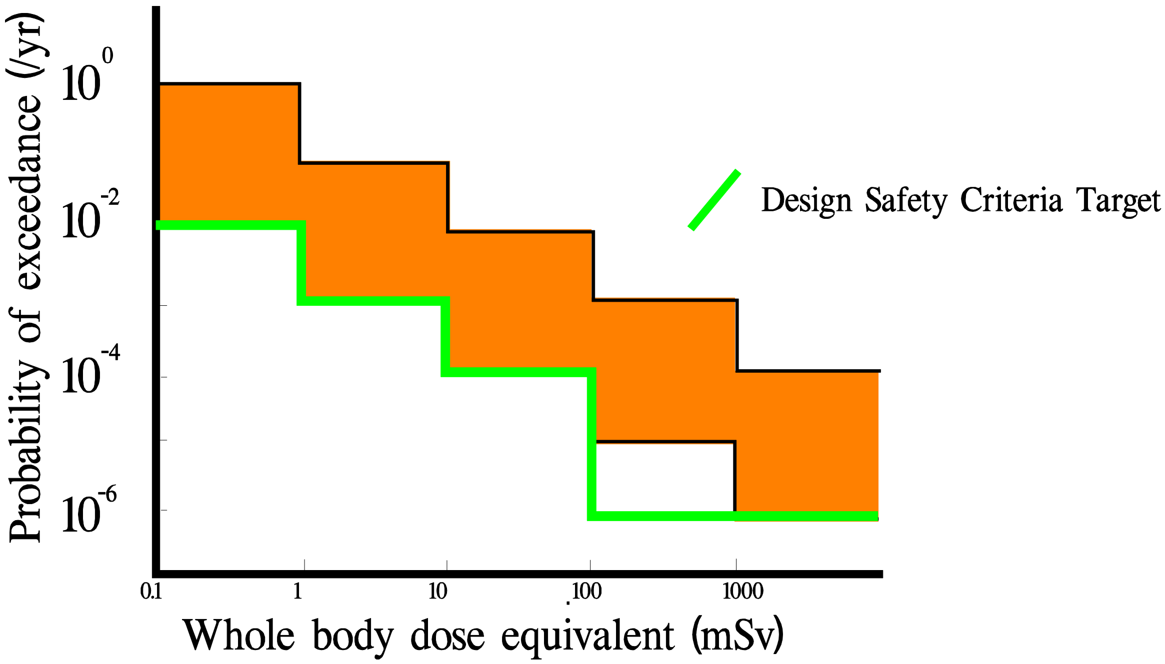

FIG. 1. Comparison between SAP and DSC criteria for Radiological Releases.

Sizewell 'B' is based on the US Standardised Nuclear Power Plant System (SNUPPS), two examples of which are successfully operating at Wolf Creek and Callaway. Changes to the design were necessary to meet UK safety requirements. Many of these are deterministic rather than probabilistic in nature: for instance the application of the single failure criterion to hazards leads to a requirement for four way segregation at power and hot shutdown. However, probabilistic/reliability targets had a strong influence on the design, which will be briefly reviewed here.

The probabilistic targets in the DSCs were applied in the design phase in two stages. As far as the design of systems was concerned, the most important of the probabilistic target was the one that sets the frequency target for uncontrolled releases for single accidents. This criterion sets the level of protection required for any given fault, since the product of the initiating fault frequency and the probability of failure to control the accident should be less than 10^{-7} per reactor year. In order to limit the reliance, which the designer puts on redundancy to achieve the reliability of protection required, the DSCs put limits on the reliability, which can be claimed for a redundant system, due to the possibility of common cause failures (CCF). The limit generally applied at the design phase for systems with active components is 10^{-4} per demand. This leads to the requirement that for frequent faults (i.e. frequencies $\geq 10^{-3}$/yr) two diverse means of protection are required. The application of this to the SNUPPS design led to the provision of the following additional diverse safety features.

– a diverse secondary protection system (SPS);
– a diverse fast acting shutdown system - the emergency boration system (EBS);
– diverse and redundant auxiliary feed water systems;
– steam turbine driven emergency charging system (ECS).

In addition to these changes to introduce diversity other changes were made to improve the performance and reliability of the ECCS and other systems. Some of these were a result of the application of the “30 minute rule” (i.e. no operator action should be necessary for at least 30 minutes for all faults within the design basis) as well as the probabilistic targets. These changes included: -

- Use of larger accumulators;
- 4 larger capacity high head pumps capable of taking direct suction from the recirculation sumps;
- Automatic changeover to recirculation;
- 4 100% essential diesels feeding 4 separate boards;
- Use of Pilot operated safety relief valves (SEBIM) to replace PORVs;
- Secondary containment;
- 4 Essential Service Water Pumps.

Having considered the impact of the probabilistic targets at the single sequence/system level a Level 1 PSA for internal initiators at power was carried out to confirm the adequacy of the design. This analysis was reported in the Preconstruction Safety Report (PCSR) (Ref. [8]) and identified further design refinements, which included:

- Provision of two Battery Charging Diesels to give long term d.c. power for control and instrumentation in the case of extended loss of all a.c.;
- Additional, diverse, isolation provisions for the containment mini-purge isolation system.

3.2. Use of PSA to support the Public Inquiry

In parallel with the PCSR analysis, a preliminary Level 3 PSA was undertaken. This was carried out by Westinghouse and the National Radiological Protection Board and was presented as part of the CEGB’s evidence to the Public Inquiry. This PSA (WCAP 9991 (Ref. [9])) was based on the same initiating fault groups used in PCSR but also considered two beyond design basis initiating faults (BDBIFs): reactor pressure vessel (RPV) failure and the Interfacing Systems LOCA (V-sequence). The PSA evaluated, not only the core damage frequency, but also a range of measures of individual and societal risk.

Although not intended to influence the design the review of the results did lead to some design changes. The results showed that the risk of death was dominated by the V-sequence and additional isolation valves were put into the RHR suction lines to reduce the V sequence frequency. The analysis of containment response concluded that the presence of water in the reactor vessel cavity at the time of RPV melt-through was beneficial from the point of view of providing protection against basemat failure. As a consequence, changes were made to the final design to engineer in passive features, which would ensure that this was the case. In addition to increase the reliability of containment safeguards the specification of one of the backup containment cooling systems was changed to cover severe accident conditions,

The presentation and discussion of the PSA at the Public Inquiry was extensive and formed the basis for the exploration of plant related issues, accident progression phenomenology, fission product behaviour and off-site consequences. The PSA was valuable as a means of putting these issues into context. The use of numbers was also seen as valuable in the context of a forum with included non-technical as well as technical assessors. The use of “engineering judgement” to assess the adequacy of a case is valid but by definition is only open to those with the requisite engineering knowledge. PSA numbers in principle give the non-engineer a yardstick but the interpretation of that yardstick is important; particularly where it involves such small numbers. In general in seeking to demonstrate low numbers the designer is seeking assurance that the sequence will not occur but in doing so is using a technique that will not rule it out absolutely.

3.3. The use of PSA during the pre-operational phase

The probabilistic analyses carried out to support the POSR were seen as having a number of functions. The main objectives were:

(i) to provide evidence that the design has conformed to the ALARP principle;
(ii) to demonstrate that the fundamental aim on which the DSC were based is met i.e. risk of death to any individual member of the public is $<10^{-6}$/yr.

In addition to providing confirmation of the adequacy of the final design the analysis was expected to provide a number of other benefits including

- an input into the optimisation of operating and accident management procedures;
- an input into the optimisation of Technical Specifications.

In clearing the PCSR a number of commitments were made with regard to the scope of the POSR fault analysis. These included:

- a comprehensive treatment of all initiating faults including those outside the design basis;
- detailed modelling of support systems;
- extension of the analysis to include internal and external hazards;
- consideration of all power states of the reactor including shutdown;
- more detailed treatment of human factors including operator errors of commission.

PSAs in general have not formed part of the formal safety analysis report for nuclear plants. For the Sizewell ‘B’ POSR this is not the case. As a result of this with the attendant requirements for the analysis to be complete and fully justified, the “PSA” has been driven away from being a “best estimate”. The scope of the analysis has been far wider than any other PSA and the need to fully justify the analysis has led to the use of bounding assumptions rather than best estimates. For this reason the analysis is generally referred to in the POSR as a probabilistic “Fault Analysis” to distinguish it from a “standard” best estimate PSA.

Conservatisms have been introduced into the PSA in a number of different ways. In particular, in many areas bounding assumptions and conservative biases have been used. In part, this has been the approach adopted to fully justify the assumptions (noting that in a regulatory arena the emphasis on justification tends to be on showing that you have not been optimistic) and partly to reduce the analysis requirements to a manageable level.

The transient analysis carried out for the POSR exemplifies this. A comprehensive event tree analysis was carried out to identify all sequences, which could lead to radiological releases (both within and outside the design limits). This identified a very large number (~5000) of design basis faults (DBFs), which were bounded in a two-stage process to produce about 90 Bounding Limiting Design Basis Faults (BLDBFs), which were analysed. These faults then, in effect, set the PSA success criteria. Conservatism was introduced in two ways.

Firstly, the BLDBFs, which characterised a group, and therefore set the success criteria for it, involved multiple failures, whereas the most frequent faults in the group would only involve single failures and could be coped with more easily. Ross (Ref. [11]) presents an example of this where a group of sequences with frequency of about 3 x 10^{-4}/yr is characterised by a BLDBF with an extremely low frequency.

The second way in which conservatism is introduced is in the analysis itself. Since the BLDBFs provide a comprehensive bounding of the design basis, they are also used in the POSR as part of the demonstration of the robustness of the design in much the same way as far less complex faults are used in Chapter 15 of US Safety Analysis Reports. The analysis has therefore been carried out with traditional design basis assumptions. These include conservative boundary conditions and data, which extends to the use, in some cases, of unphysical combinations of parameters such as start of life pellet-clad gap combined with end of life decay heat.

Although transient analysis has been used as an example here, the use of bounding assumptions and pessimistic parameters permeates safety analysis reports produced in a regulatory arena and has had a strong influence on the Fault Analysis for Sizewell ‘B’.

The adoption of a conservatively biased approach had a strong influence on the approach to uncertainties. At the Sizewell ‘B’ Public Inquiry there had been evidence presented by a number of parties on the treatment of uncertainties. Both NII (Ref. [12]) and CEGB (Ref. [13]) argued that quantitative statistical uncertainty analysis was inappropriate where the analysis was conservatively biased. However, the identification of key uncertainties and the use of sensitivity studies to establish the sensitivity of the overall results to these were seen to be useful.

The commitment to include greater detail in the PSA led to an extensive fault schedule (more than 180 initiating faults were considered) and safeguard schedule. This, allied with the requirement for more detailed support system modelling and the need to address all operating states, has led to a very extensive and detailed fault tree analysis. In addition to the comprehensive coverage of initiating faults the analysis has covered about 80 BDBIFs, which include the incredible initiating faults such as RPV and steam generator failures. This should be compared with “typical” PSAs, which consider 2 such faults: RPV failure and the V sequence. The analysis of Internal and External hazards started from a comprehensive list of about 60 possible hazards. Some of these were eliminated by screening or bounding; the remainder were quantified. All the fault groups discussed above were quantified at all power states and at shutdown. Contributions were also included from non-reactor core sources of radioactivity. These included contributions from the radwaste plant and the fuel route.

One of the commitments made at the PCSR stage was to include human error in the analysis. Human error is already implicit in the failure data from which the initiating fault frequencies and component reliabilities have been derived. Operator error has also been modelled in the fault and event trees but it was recognised that there may be (albeit a few) contributions which could not be modelled in this way. A further “direct estimation” route extends the normal fault tree analysis average. This is largely based on reviews of Operating Instructions to look for the potential for operator error to lead to plant damage not already covered in the analysis.

As the analysis proceeded it became apparent that the many bounding assumptions were tending to grossly pessimise the results. Since it would probably be a totally intractable task to carry out a best estimate analysis, from first principles, with the level of detail required, an iterative approach was adopted in which the results of the analysis using bounding assumptions were reviewed to identify the worst of the pessimisms. Where necessary, additional analysis was then undertaken to underwrite the reduction in the degree of pessimism and the overall results were modified accordingly. However, the analysis is still significantly more conservative than would be normal for a "standard" PSA. This was revisited in the production of the "Living PSA" (see section 4).

3.4. Sizewell 'B' Fault Analysis Results

The total predicted frequency of conditions, which potentially lead to a dose of > 100 mSv to someone at the site fence, from all initiators at all power states is about 10^{-5}/yr (Ref. [14]). The contribution to the maximum individual risk of death from all accidents is about 10^{-7}/yr. The relative contributions to individual risk and core damage frequency are shown in Table I, based on the POSR indicative analysis (i.e. the revised pessimistic analysis, discussed above).

TABLE I. CONTRIBUTIONS TO THE POSR INDICATIVE ANALYSIS

Contributor	% Contribution to		
	Individual Risk	Core Damage Frequency	Large Release Frequency
Internal Initiators at Power	9.5	26.9	11.5
Internal Initiators at shutdown	12.8	31.9	40.6
Hazards at Power	2.4	2.5	5.0
Hazards at Shutdown	6.7	21.6	32.8
BDBIFs at Power	13.9	4.8	5.0
BDBIFs at Shutdown	14.6	5.4	4.0
Operator Error	2.1	6.9	0.7
Ex-reactor faults	<0.1	-	0.2
Design Capability Faults	38.0	-	-
Total faults at Power	56	40	22
Total faults at Shutdown	43	60	78

The first point to note is that the analysis is generally pointing to the importance of faults at shutdown. Indeed this is consistent with what was observed from the studies of internal initiators at power and shutdown for French plants (Ref. [15]). In going from the French three loop 900 MW plants to the four loop 1300 MW plants and then to Sizewell 'B' ones sees both reductions in the predicted core damage frequencies for the more modern plants and a tendency for faults at shutdown to become relatively more important. However, the Sizewell 'B' POSR analysis also showed the importance of the shutdown states also appeared to apply to hazards. The shutdown analysis results have led to the definition of Technical Specifications to cover shutdown states for Sizewell 'B'.

In terms of individual risk the analysis seems to indicate that it is those faults within the design capability of the plant (i.e. those not associated with core damage), which give the largest single contribution to risk. These success states are not normally included in PSAs. Although this conclusion may well be true it needs to be treated with some caution since these design basis faults have been treated even more conservatively than the others. For instance they have been identified in terms of the limiting ERL for evacuation (as specified by the DSCs), which is generally the dose to the thyroid. However, in evaluating the individual risk they are conservatively associated with the release that would give the equivalent whole body

ERL. In general this alone will result in an over-estimate in the risk from these faults by at least a factor of 3.

This illustrates a general problem associated with the comparison of different aspects of the overall assessment. The Tolerability of Risk Report (Ref. [5]) sets the broadly acceptable level for individual risk at the 10^{-6}/yr level (as does Nuclear Electric) but interprets this as covering normal as well as fault operation. For Sizewell 'B' the contribution from normal operation can be estimated from the dose to the "critical group" and does give a total below the 10^{-6}/yr level. However, the normal operation estimate is, again, extremely conservative, both because of conservatisms in the analysis of the possible levels of routine discharges and also because the critical individual whose risk is estimated does not actually exist. He is himself a result of bounding possible behaviours, some of which may involve being in two places at the same time.

Despite the conservatisms in the analysis the Sizewell 'B' fault analysis confirms the very low level of risk associated with the plant.

In addition to providing an input into the design and safety substantiation of the plant the PSA was used to assist in the development of the Technical Specifications for the plant. These were based on the draft MERITS (Methodically Engineered, Revised and Improved Tech. Specs.) Tech Specs (Ref. [16]). However, Sizewell B had equipment, which was not found in other plants and a safety case, which was more extensive in its coverage. The safety case was used as the starting point, and sensitivity studies, using the PSA model, were used to determine, on a risk informed basis, both whether new plant should be included in Tech Specs or lower level documentation and to refine the action times to control both average and point in time risk. This did not include, at this stage, a review of whether some of the items included in standard Tech Specs on deterministic grounds could be justified on a risk basis.

4. USE OF PSA DURING OPERATION

A great deal of effort went into production of the licensing PSA and its use in both cost benefit/ALARP justifications as well as in the refinement of Tech Specs indicated that it had a potentially valuable role in the support of operation. In particular it could have a role in:

- Plant Modification assessment
- Technical Specification modification
- Procedure modification
- Safety assessments
- Maintenance optimisation.

The POSR PSA analysis was undertaken using an all fault tree approach (i.e. functional fault trees and system fault trees). Event tree analysis was used primarily to identify initiating faults for the analysis of within design basis faults. The choice of the fault tree approach was influenced by a number of factors but one important one was associated with the recognition of the need to model support state dependencies. At the time that the decision was being taken on the analysis approach (early to mid 80s), event tree - fault tree packages, with true fault tree linking, were less readily available than they are now. The all fault tree approach was adopted as one that could handle the detailed modelling of support system dependencies. The platform used was capable of handling the very large fault trees but was relatively slow running.

The decision was therefore taken to produce a living PSA model by both, re-platforming the model, and reducing the levels of conservatism in some aspects of the modelling (e.g. the definition of some of the success criteria). This later aspect sought to provide a more rigorous justification of the work done to produce the POSR "Indicative" analysis. After reviewing the alternatives a modern linked event tree, fault tree package was selected. The use of such a package, as well as providing rigorous analysis was also seen to be advantageous in presenting results to non-specialists. It had been found at the public inquiry that event trees were regarded as being more intuitively understandable than fault trees.

The Living PSA (LPSA) model has been produced and approved for use in support of a range of operational activities. It also supersedes the PSA analysis in the Station Safety Report. Figure 2 shows a comparison between the POSR Pessimistic, Indicative and LPSA results. This confirms that the POSR pessimistic results did indeed distort the balance between faults at power and those at shutdown. The judgements made in the indicative analysis were a better reflection of the balance (though the more detailed analysis identified an increase in the contribution from faults at shutdown rather than a decrease). In addition the LPSA analysis showed that the POSR analysis overestimated the importance of hazards at shutdown. In the LPSA analysis it is hazards at power that dominate. The LPSA analysis also showed that even more of the operator error contributions could be incorporated into the fault trees so the direct estimate contribution is significantly reduced.

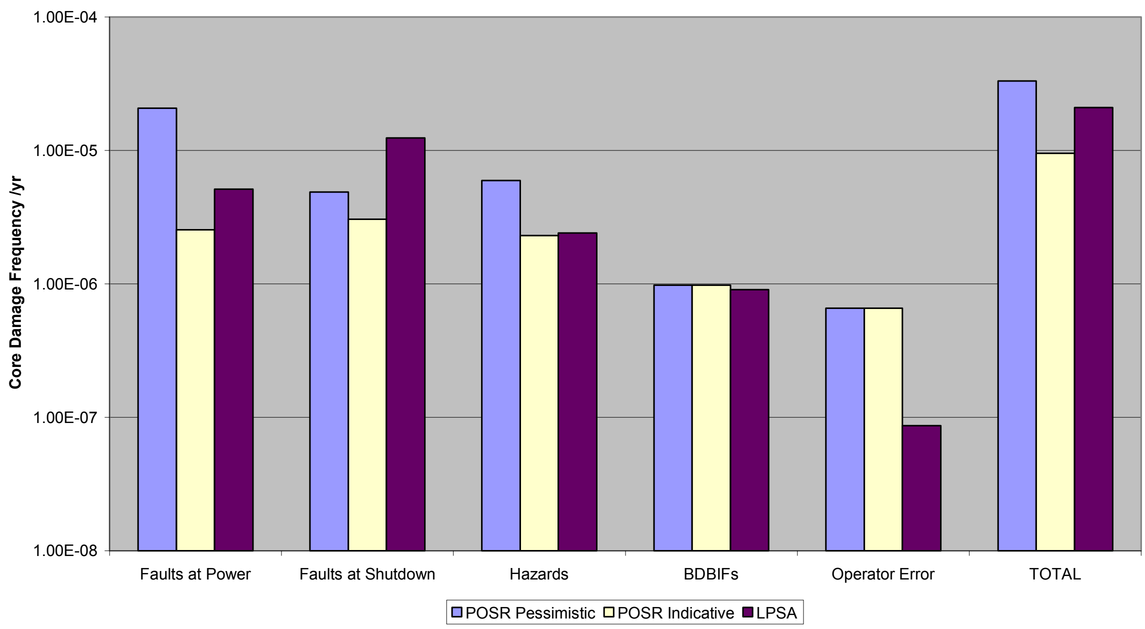

FIG. 2. Comparison between analyses.

5. LESSONS LEARNED FROM THE APPLICATION OF PSA TO SIZE WELL 'B'

The Sizewell 'B' probabilistic analysis has been very extensive and has been applied in a number of areas. Some of the lessons learned are briefly discussed below.

5.1. The Use of PSA as a Design Tool

Sizewell 'B' has, we believe, successfully demonstrated the use of PSA as a design tool. It offers a comprehensive and systematic way of reviewing the design. From the point of view of the initial design process, probabilistic targets usefully complement deterministic design rules.

The PSAs have also been used as a means of confirming or "validating" the design. There is sometimes a desire to use PSA results to compare different designs. A detailed comparison on a similar basis may be useful since the PSAs should aid the understanding of the individual designs. Differences in scope and assumptions make simple comparisons of "bottom line" numbers potentially misleading. The use of a "standard" PSA specification help this process, but the main benefit to be gained is generally from an understanding of your own design.

The POSR fault analysis has been used to examine whether the design improvements incorporated into Sizewell 'B' were worthwhile. The effect on the core damage frequency of removing some of these features from the design has been estimated. The removal of three systems was considered:

– Secondary Protection System
– Emergency Boration System
– Emergency Charging System

If these were not present using the POSR data and assumptions the core damage frequency would increase by a factor of about 40, which is roughly consistent with the comparisons with the results of earlier PSAs, performed on US plant. However, the size of the benefit is very sensitive to the data and assumptions used.

One question, which is often asked, is to what degree was the PSA responsible for the increase in complexity of the Sizewell B plant? This is rather difficult to answer directly. Generally the extra plant added to the Sizewell B design were the result of *deterministic* rather than *probabilistic* requirements. However, some of the deterministic requirements had a probabilistic basis. The PSA was also used to justify not adding systems. For instance the PSA was used to demonstrate that, having already upgraded the containment overpressure protection, the addition of a filtered venting system would not be ALARP.

5.2. The Use of PSA in a Regulatory Arena

In principle, all the benefits that can be derived from the use of PSA in the design process, in terms of an improved understanding of the plant, should be applicable in the licensing process. However, this is not always as simple to achieve once the PSA becomes integrated into the formal safety case unless its particular "best estimate" role is recognised. In the Sizewell 'B' case there has been a tendency for the needs of the probabilistic analysis to complicate the design basis analysis and for the use of bounding licensing assumptions in support of the design basis analysis to make the PSA overly conservative.

This is not a unique problem; US utilities expressed similar concerns with respect to the regulatory use of the individual Plant Examinations (Ref. [17]). The problem, real or imagined, is associated with the application of the very cautious "licensing" assumptions to what is intended to be a best estimate analysis. This seems to have been recognised in the new SAPs (Ref. [6]) where NII have attempted to separate the requirements for design basis and PSA analysis. However this was not the case for the POSR analysis.

From a utilities point of view the benefits of a PSA are that it gives a much greater understanding of the types of failure sequence that can lead to a significant release and hence how to guard against it. It is important that this understanding should be based on realism rather than be distorted by unnecessary conservatisms (though some conservatisms will inevitably be needed to make the analysis manageable). This is clearly illustrated by the differences in balance between faults at power and shutdown seen in the pessimistic versus the indicative and LPSA analysis.

The detail present in the model does allow the role of systems to be put in the overall context of their impact on the safety of the plant. For instance, the PSA has been used to examine the dependence on the Primary Protection System (PPS) reliability. This has demonstrated that because of the provision of a Secondary Protection System the risk from the plant is not unduly sensitive to the PPS reliability.

This ability to put “safety issues” into a risk perspective is valuable in the context of plant licensing. In particular the PSA is a useful tool to examine compliance with the ALARP principle. The PSA can be used to identify the principle contributors to the various measures of risk both in terms of the failure sequences and the most significant plant failures. This particular feature was used to refine Technical Specifications and Operating Procedures, as discussed above.

5.3. The use of PSA to refine Operating Instructions and Accident Management

The POSR fault analysis made relatively few claims for “accident management” actions, but as part of the ALARP reviews the potential role of operator recovery actions was identified. A comprehensive set of Station Operating Instructions (SOIs) had been produced and in most cases these had already covered the required recovery actions. However a number of additional actions were identified. Review of the level 2 PSA results identified a number of potential severe accident Management measures (Ref. [18]), which have now been incorporated into the SOIs (which cover severe accidents as well as design basis sequences).

5.4. Some thoughts on the use of PSA in support of performance optimisation

This is an area that is starting to be pursued in response to the changing climate NPPs are finding themselves operating in. The focus on safety remains unchanged. However most plants now find themselves operating in an increasingly aggressive commercial climate. The consumer wants cheap energy, safe production and shows an increasing regard for environmental protection. Politicians tend to promise all three, but the means of achieving this is not clear, and the emphasis varies with time. PSA offers the potential for performing optimisations of safety, performance and environmental impact, in that it should be possible to arrive at a situation in which all are simultaneously treated. This will, however, always tend to result in a balance being struck. There are a number of factors, which need to be taken into account in establishing a correct balance.

5.4.1. How to establish a level playing field

To arrive at a correct balance requires all the elements to be compared on a like for like basis. This is possible in principle and is most commonly achieved by using cost as the common currency. It is possible, in principle, to ascribe costs to safety detriments and to adverse environmental impacts. However this gives rise to presentational difficulties in that it is perceived that you are allowing “commercial issues” to compromise safety. The use of a quantitative approach allows you to explicitly deal with such issues; the relative weightings can be adjusted to reflect public concerns. (The ToR document (Ref. [6]) applies very large risk aversion factors in setting the targets for nuclear plants relative to major conventional plant risks.) However, this strength is also a potential weakness, in that people often seem to prefer that such decisions are implicit. In addition different “regulators” have responsibility for each aspect of the problem and they are mainly charged with optimising that aspect. The need for a balanced approach is recognised in principle by the Health and Safety Executive in a recent discussion document (Ref. [19]).

5.4.2. Completeness and Human Factors modelling

There has always been a strong emphasis on completeness in formulating the Sizewell B PSA. This partly arose from the fact that we were using a "bottom line" demonstration of individual risk as the most fundamental criterion. This led to the requirement to include beyond design basis initiating faults, operator errors of commission etc. However one aspect relating to operator error is probably not modelled very well and that is the beneficial human performance. We model the operator as a source of error, the maintainer as a source of common mode failure but rarely, except in the long term, model the operator as a robust line of protection. Indeed the intention in the design was wherever possible to provide automatic protection rather than rely on the operator. On the other hand, plants invest a great deal of time and money in improving human performance by for instance adopting and implementing the INPO/WANO Performance Objectives and Criteria. It is not clear that the PSAs reflect the benefits to be accrued from this approach. This would seem to limit efforts on safety improvement to plant design features, which does not seem reasonable.

5.4.3. The role of Deterministic Criteria

As was noted above, deterministic criteria are generally needed to simplify both design and analysis. They are, (or should be), surrogates for risk criteria but in the past there has been a reluctance to use risk based analysis to modify or simplify them. The more widespread use of risk informed methods should help here but it does need to recognised that PSAs do have limitations, which need to be accounted for in their application.

6. CONCLUSIONS

The application of PSA to Sizewell B has demonstrated that it is a powerful tool with potential for future use. Its strengths and limitations as a tool need to recognised by both users and regulators. It is not a fully mechanistic means of ensuring design safety, but is an important aid to decision making. It also has the potential to allow risk judgements to be taken in conjunction with commercial and environmental issues.

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Annex 15

Cost and risk reduction using upfront licensing in Canada

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Abstract. The paper summarizes the use of "up-front" licensing in Canada – how licensing requirements are defined, and met – in advance of a project commitment. The approach to licensing in Canada has allowed flexibility in development of new designs. Since licensing was originally risk-based, and current regulatory policy allows cost-benefit considerations as part of the decision making, risk can be and should be used in novel circumstances as a licensing tool. Since the licensing framework is non-prescriptive, innovative approaches to design can be introduced and dispositioned without changing the legal structure. This flexibility has been used in several up-front licensing reviews: a small urban heating reactor, repeat CANDU® 6 generating station units, and the single unit CANDU 9 generating station. In the future we expect to apply it to advanced designs, as an essential part of risk reduction and customer confidence in the product. The important lessons learned in Canada include:

- Up-front licensing is essential to reduce the risk of licensing-related delays once a project has been committed. It requires a significant investment in time and effort from both the designer and the regulator;
- The most effective scope for up-front licensing is for the regulator to thoroughly assess novel concepts, test the design against changed domestic requirements, and follow-up on known difficult areas; and for the designer to ensure foreign requirements are incorporated. There is little benefit in certifying the design in detail;
- Although it would be satisfying to have legally-binding certification, in the end there can be no legal obligation on the regulator, and agreement is pursued on the basis of good faith that the regulator will not make arbitrary decisions and that the designer will meet agreed targets or requirements;
- In almost all circumstances, issues will arise that are beyond the current 'rules', however expressed. Rather than rushing to create new rules, one reaches a sensible conclusion. The conclusion becomes precedent; precedent becomes practice; practice becomes a regulatory guide;
- PSA, and its companion, cost-benefit analysis, play an important role in up-front licensing, as a way of casting decisions in an objective framework;
- Some form of international licensability or generally-accepted framework will be increasingly required to support nuclear power plant projects and to avoid continued re-licensing, overlapping reviews and non-standard requirements.

1. INTRODUCTION

The drivers for cost and risk reduction on new nuclear construction have, if anything, become more acute over the last decade. Originally introduced as a reaction to the overspending and lateness of nuclear power plants in the 1980s, the opening of electricity markets to competition has meant that the pressures to complete a project on time and within budget are even stronger. In fact overcoming such risks is essential if nuclear power is to have a future.

Licensing has, fairly or unfairly, been identified as a source of such risks. Whether the licensee or the regulator was 'at fault' is almost impossible to agree on: what is more certain is that requirements were not clearly enunciated at the beginning. Indeed, if the regulatory requirements are too onerous, it is better to find out at the beginning, since the project can then be discontinued with little penalty, and the reasons are then clear.

Definition of licensing requirements at the beginning, or 'up-front' – and indicating how they will be met, up front – is the subject of this paper. The emphasis will be on what the author knows and has used: the Canadian approach and experience.

2. LICENSING APPROACH IN CANADA

2.1. History — risk-based origins

Canada's approach to accidents began with the accident to the NRX pressure tube reactor, in 1952, Refs. [1-2]. This spurred an early interest in both the frequency of accidents, and the nature of protective systems, particularly their separation from the process systems which normally control the station.

These ideas were enunciated in a paper in 1959 by Ernest Siddall [3], then with the Reactor Research and Development Division at CRNL. He took as a safety goal that the risk from nuclear power should be five times lower than the risk from coal power, which was then the alternative in Ontario for future electricity generation. He compared the two power sources on the basis of prompt fatalities, including the front-end fuel cycle for both. From this he derived a target for a remotely-sited nuclear power station of 0.2 deaths/year on average. This risk was felt at that time to come mainly from the catastrophic accident, as described in the U.S. WASH-740 report. Assuming these results applied equally to a Canadian reactor, he produced a set of maximum event frequencies and safety system unavailabilities to be used as design targets, as follows:

LOSS OF COOLANT	**One in 50 years (0.02 / ry[1])**
LOSS OF POWER CONTROL	**One in 16 to one in 160 years, depending on severity (0.06 – 0.006 / ry)**
SHUTDOWN SYSTEM UNAVAILABILITY	**One in 500 tries (0.002 yrs / yr)**

In simple terms, a catastrophic accident such as postulated in WASH-1400 could occur only if a process system failed (pipe break, loss of power control) *and* the shutdown system failed. One could estimate the frequency of the catastrophe by multiplying event frequencies by safety system unavailabilities (e.g., [1 per 50 years] times [1 in 500 tries], or once in 25,000 reactor years). Now this is only possible if the systems are sufficiently independent, i.e., if there are no major cross-links between the initiating event and the mitigating system. This philosophy of separation (logically and physically) between process and safety systems has been one of the hallmarks of CANDU® design from then until today.

A similar approach was followed in a paper in 1961 by G.C. Laurence [4], who was then director of the Reactor Research and Development Division at CRNL, and who later became President of the Atomic Energy Control Board (AECB), the regulatory agency of Canada (now Canadian Nuclear Safety Commission, or CNSC). He took as a **safety goal** 10^{-2} deaths per year from nuclear power plant accidents, a factor of 10 lower than Siddall's, with the justification that this was far better than in other industries. With remotely-sited plants, which were then the only locations being considered, a disastrous accident would cause fewer than

[1] ry = reactor operating-year

1000 early deaths, so the frequency of such disasters must be held to less than one per 100,000 years. Such a disaster could occur if we had a simultaneous failure of **all** of: a normal process system (such as the reactor power control system), a protective system (emergency core cooling or shutdown) and containment. From this he derived the following design targets:

Process failures	**One in 10 years**
Protective System Unavailability	**One in 100 demands**
Containment System Unavailability	**One in 100 demands**

The frequency of process failures seems rather undemanding - for example no utility would tolerate a plant with a predicted large LOCA frequency of one every 10 years[2]. The numbers should be looked at minimal requirements for public safety, not risk estimates. There is no point setting targets if performance cannot be measured. Thus the numbers were chosen **large enough to be demonstrable individually by experience or testing in a few years of reactor operation**.

These ideas were applied in the design of Canada's first demonstration power reactor - the Nuclear Power Demonstration (NPD) Reactor. Its 1961 Hazards Report used higher unavailability for shutdown, and did not credit containment. It also assessed the dose to the public from less severe accidents than disasters, using as a figure-of-merit a "once-in-a-lifetime" emergency dose. For Iodine-131, for example, this was 25 rad.

The 1962 Safety Report for the 200 MWe Douglas Point nuclear reactor was perhaps the fullest flowering of the overall risk-based approach. The safety goal – proposed by the designers – was that the risk of death to any member of the public be less than 10^{-6} per year, a factor of 10 less than that for NPD. The target risk for injury was taken to be 10 times larger than the risk of death, in the same ratio as experienced in other industries. The breakdown by frequency was similar to that for NPD, with some allowance for the lower frequency of large pipe breaks. Included in this risk evaluation was a quantification of the effects of a major accident on the operating staff. The Safety Report consisted of a systematic listing of all identifiable events, an evaluation of their frequency, and a calculation of their consequences in terms of dose. Again, separation was assumed to be achieved by careful design practice. Note in addition the increasing requirement for nuclear not just to be safer than coal, but to be orders of magnitude safer. This was partly due to the fact that it was a new technology and the “increased safety” seemed achievable, and partly to cover uncertainties. However this idea did result in an erosion of the rationale for optimizing safety across industries.

Note that these numerical goals for individual risk correspond very closely to the implied targets for severe core damage ($<10^{-4}$ / yr) and for large releases ($\sim10^{-6}$ / yr) discussed by the NRC in the USA, and proposed by F.R. Farmer in his pioneering work in the UK on risk-based safety analysis.

2.2. The single/dual failure approach

In 1967, F.C. Boyd of the CNSC[3] laid the ground rules for the deterministic licensing guidelines, under which all operating CANDU plants up to Darlington have been licensed.

[2] Hancox & Meneley in 1982, describe the plant-protection safety requirements for CANDU (Ref. [5]).

[3] To avoid confusion, we will henceforth use the term CNSC even when the historical context would require AECB.

They showed evidence of their risk-based origins, but collapsed the spectrum of possible accidents into two broad categories: **single failures**, or the failure of any one process system in the plant; and **dual failures**, a much less likely event defined as a single failure coupled with the unavailability of either the shutdown system**,** containment, or the emergency core cooling system - the so-called *special safety systems*. (This single failure is an assumed *system* failure, and is not related to the same term used for Light Water Reactors to describe a random *component* failure additional to the initiating event). For each class, a frequency and consequence target was chosen that designers had to demonstrate were met. In addition, to deal with the siting of a reactor (Pickering A) next to a major population centre (Toronto), *population* dose limits were defined for each class of accident.

The single-dual failure guidelines were finalized in 1972 by D.G. Hurst and F.C. Boyd of the CNSC, Ref. [6]. The guidelines were as follows in Table I:

TABLE I. DOSE/FREQUENCY GUIDELINES

Accident	Maximum Frequency	Individual Dose Limit	Population Dose Limit
Single Failure	1 per 3 years	0.05 Sv 0.03 Sv thyroid	10^2 Sv 10^2 Sv thyroid
Dual Failure	1 per 3000 years	0.25 Sv 2.5 Sv thyroid	10^4 Sv 10^4 Sv thyroid

Although the single-dual failure approach was a move away from the early risk-based days, it still retained some risk roots (event classes and dose limits based on frequency).

2.3. Current practice

2.3.1. Deterministic requirements

To address some of the deficiencies in the single-dual failure methodology, *but still within the design-basis accident approach*, the CNSC issued document C-6 in June 1980 [7]. This retained the concept of several classes of events, five in this case, but with important differences:

1. Although the classes represented decreasing event frequency, assignment of events to the classes was done *a priori* by CNSC staff, based on their estimate as to the likelihood of the event. The assignment had a conservative bias, with the result that an analysis done in the framework of C-6 could give a distorted picture of 'real' safety. Also by assigning events to a class, the document removed from the designer some of his incentive either to show that an event was indeed less frequent, or to make changes to decrease the frequency. Indeed, the list of events is highly design-specific, and might not be sensibly applied to future plants - a significant limitation as new generations of CANDU are developed.
2. To avoid the appearance of any increase in the maximum "permissible" dose in the new system, the CNSC set the maximum dose for the most infrequent class at 0.25 Sv whole body: in other words, events *less* frequent than the traditional dual failure were not recognized in terms of increased allowable doses.

3. Since the dose from each event was required to be less than a given value, there was no need to sum over the events to arrive at a risk estimate.

The limits were as follows in Table II:

TABLE II. DOSE/CLASS LIMITS FROM CNSC DOCUMENT C-6 REV. 0 (1980)

Event Class	Reference Dose Limit, Sv Whole Body	Thyroid
1	0.0005	0.005
2	0.005	0.05
3	0.03	0.3
4	0.1	1
5	0.25	2.5

C-6 can best be viewed as a deterministic approach, despite its growth from two to five classes. This document is being revised to incorporate experience.

2.3.2. Probabilistic requirements

It is obvious from the brief history above that Canada used probabilistic methods very early on in design, and continued to use them for reliability calculations of special safety systems. However their use in accident analysis was eclipsed by the single-dual failure approach. As experience was gained with the latter, accidents it did *not* capture became a concern, and it was once again supplemented by probabilistic analyses – concentrating first on the effects of support system failures, and later extending to (now)-conventional probabilistic safety analysis (PSA).

While there is no formal regulatory requirement in Canada for a PSA, in practice it is both needed (to demonstrate that all accidents have been identified – one of the requirements of C-6) and expected, and an applicant would be unlikely to get a licence for a new plant in Canada without a comprehensive PSA.

2.3.3. Cost/benefit policy

Until recently, there has been no formal cost-benefit policy applied to CNSC regulations. Cost-benefit arguments have nevertheless been made, and were dispositioned on a somewhat *ad hoc* basis. Recently, however, as a result of the Canadian government's directives on regulatory activities in all fields, the CNSC has issued a Cost Benefit Policy [8], which states that:

"The Canadian Nuclear Safety Commission recognizes that compliance with its decisions and orders entails social and economic costs that are borne by licensees and others who are subject to its control, and by other Canadians. Accordingly, the Commission's decision-making processes include the opportunity for affected persons to be heard and for others to participate. The Commission also recognizes that consultation is an important component in the development of its regulatory documents. "*It is therefore the policy of the Commission that:*

- *When conducting a proceeding for purposes of a decision under the Nuclear Safety and Control Act that involves a licence or an order, the Commission or its designated officers will consider relevant information on costs or benefits that is submitted by a person who is participating in the process,*

- *When conducting consultations on a draft regulatory standard or a draft regulatory policy, the Commission will take into account, when fixing the deadline for submission of comments, the time that may be required for the preparation of submissions on the costs and benefits related to the proposed standard or policy,*

- *When receiving or considering any relevant information on costs or benefits that is submitted in relation to a decision involving a licence or order, the Commission or its designated officers will be governed by the following principles:*

  - *Information on costs and benefits is only one factor that may be considered in making "regulatory decisions" or taking "regulatory actions" under the Act, and does not displace legal requirements and other valid regulatory considerations*

  - *The information on costs or benefits may be quantitative or qualitative in nature*

  - *Consideration of the information on costs or benefits may be quantitative or qualitative in nature."*

The key aspects of this policy are that the onus is on the proponent to make the cost/benefit case, and that cost/benefit considerations are not the only governing factor in a decision. As of this writing, the policy has not been exercised, although it its use is expected to begin in the near future.

2.4. Legal basis – prescriptive vs. non-prescriptive

It is clear from the above discussion that the Canadian approach is non-prescriptive: that is, the **regulator states safety goals which designer/operator must meet but does not prescribe how to meet them**. This gives a large amount of freedom in negotiating up-front licensing, which we describe below, because there are few legal barriers to innovation. In this respect the Canadian approach is similar to that used in the U.K. and dissimilar to that used in the U.S. Table III below gives a simple comparison between the U.S. and the Canadian approach to licensing.

TABLE III. PRESCRIPTIVE AND NON-PRESCRIPTIVE REGULATION

U.S.	Canada
Many vendors, many different designs	One vendor, one base design
Legal-oriented	Consensus-oriented
Prescribes overall requirements plus specific acceptance criteria and how to do design	Prescribes high-level acceptance criteria; onus is on the designer to justify the design
About 6 binders of detailed laws (Code of Federal Regulations)	About 100 pages of laws

As an example, we compare the requirements to ensure a coolable fuel geometry after a LOCA:

U.S.	Canada
U.S. 10CFR50 Section 46(b(1) "The calculated maximum fuel element cladding temperature shall not exceed 2200^{o}F"	Canada - R-9, Section 3.2(c) "All fuel in the reactor and all fuel channels shall be kept in a configuration such that continued removal by ECCS of the decay heat produced by the fuel can be maintained..."
Prescribes not only the limit but the models used to calculate it	Describes objective; it is up to the designer to do tests and develop models to prove it is met

The comparison is illustrative: of course, in practice CNSC will not accept temperatures so high that fuel bundle behaviour cannot be predicted, and the USNRC can be persuaded to revise its requirements. In addition the non-prescriptive nature of the Canadian regulatory requirements does not imply that they are less stringent than prescriptive requirements. One example is the requirement for two fast-acting, fully-effective and independent shutdown systems in Canada versus one fast-acting shutdown system in other jurisdictions. Another example is the requirement in Canada setting a maximum unavailability of 10^{-3} yrs/yr for each special safety system, versus the single failure criterion used elsewhere. Design and operating experience shows that simple redundancy of components within a system sufficient to meet the single failure criterion can fall short of achieving the Canadian quantitative requirement for overall system availability. In other words, a prescriptive approach means that requirements are more detailed but are not necessarily more demanding from a safety standpoint.

Finally note that safety Research and Development in Canada, including development of analytical tools, is the responsibility of the proponent (AECL and CANDU licensees), not the regulator; CNSC audits the programme and the results, and also contracts smaller-scale confirmatory R&D. This again allows flexibility and speed in bringing new R&D and analytical technology into service.

3. DRIVERS FOR UP-FRONT LICENSING

Part of the response to the delays and cost overruns of nuclear plant construction in the 1980s was development of methods for "up-front licensing". It was believed that if the requirements, and the way they were implemented, in new designs were agreed beforehand with the regulators, then a large component of cost uncertainty would be reduced. The level of detail in the agreement sought varied widely, as we shall discuss later. Unfortunately the interest in up-front licensing coincided with a general downturn in the market for nuclear power, so it was exercised only in a few, but important, cases.

Some of these concerns have lessened. Over the past 40 years a great deal of experience has been gained by designers and analysts. The essentials of a safe and sound nuclear plant design are quite well understood. When undertaking a new project, whether evolutionary or revolutionary, designers are able to establish these fundamental characteristics very early in the design cycle. Similarly, regulatory staff also understand these design essentials. This greatly simplifies the process of safety design and reduces licensing risk for any new 'evolutionary' project. Established international standards for safety performance also can give the designer an early indication of what will be required of his design. Nevertheless, there remains a risk of misunderstanding between the chosen detailed characteristics of a new design and the regulator's expectations for the acceptable safety characteristics of that design.

The existing licensing framework can of course be made more efficient. Asian countries have shown the world how, even in a traditional licensing process, licensing risk can be minimized. Their licensing process encouraged resolution of all major design and analysis issues by the time of the construction permit, so that there was less risk of delay between construction and operating licences. One can, of course, view this as a form of "up-front" licensing, with the issue of the construction permit marking the end of it.

In some countries there are legal barriers to up-front licensing. A regulator may not be permitted to engage in negotiation with a proponent until there is a formal application for a construction licence, by which time project schedule pressures do not allow time for up-front licensing. Or such discussions may occur but the regulator cannot be bound to them until there is a project. Such discussions are nevertheless immensely useful, especially if they build up both knowledge and trust.

In the end, however, there are powerful reasons to undertake a formal up-front licensing process.

If there has been no recent "live" regulatory experience, up-front licensing may be essential to exercise the regulatory process before a major commitment is made. For example, in Canada, there has not been a new nuclear power plant operating license granted since 1993; a similar situation holds in the U.S. In Canada, the CNSC has tended to focus on the problems of operating plants, but has nevertheless continued to issue some Regulatory Guides which would apply mainly, or only, to new plants. Until these have been understood through use in a real application, they represent a large risk: since the words may not match the intent; since regulatory staff who know the intent may have left; and since some of the requirements may be impractical or require extensive interpretation.

If new technology is intended for use in a series of plants, up-front licensing becomes essential – for example, if further R&D or changes in design concept are needed as a condition of regulatory acceptance, the proponent needs to know before a project is struck.

As plants age, life extension becomes an attractive option. Most regulatory jurisdictions will expect at least some sort of safety assessment of the plant at the time of a request for life extension or refurbishment. Agreement beforehand on the standards against which the plant is to be reviewed, and what is to be done in case of gaps, is essential to assure the owner of a success path.

Finally most purchasing countries still require licensability of the nuclear power plant in the country of origin. It is preferable if a "real" licensed reference plant exists; if not, the next best thing is a thorough review by the host country licensing body followed by a declaration of licensability. The reason is that regulators in purchasing countries, though they may be as fully capable technically as in the host country, may not be fully aware of the vast amount of unwritten experience which can only be developed slowly as a country gains experience – a matter of knowing "why" as well as "what" and "how".

4. EXPERIENCE WITH UP-FRONT LICENSING

Canada has had at least three campaigns of up-front licensing, summarized below, which demonstrate the effectiveness of the approach for a range of plant designs and sizes.

4.1. Point Lepreau 2

In the mid-1980s, there was a possibility of building an additional CANDU 6 unit at Point Lepreau (nominal output 600 MWe), in New Brunswick – alongside the original one, which began operation in 1983. Point Lepreau 1 had taken 7½ years to complete, compared to 39 months in Japan (Takahama 3) with its one-step licensing process. By no means were all of the delays in Point Lepreau 1 due to licensing; but if licensing were not handled in advance, it could become a schedule risk.

AECL therefore set up, with the CNSC, a process for defining and resolving regulatory requirements, Refs. [9-10]. In this particular case, the process relied heavily on the fact that Point Lepreau 1 was a licensed, operating plant. The approach was to:

- define, document, and obtain agreement on the licensing requirements;
- identify and assess the impact of new requirements on plant design during the pre-project phase and implement necessary changes to the design early;
- utilize Point Lepreau 1 safety assessments to the greatest extent possible;
- perform safety analysis work early, to minimize the potential impact on project schedule;
- establish a licensing schedule which would allow a timely and orderly submission and review of safety assessments; and
- obtain a conditional operating licence at construction licence time.

Most of these steps were used in subsequent applications of up-front licensing, described below. The importance of defining regulatory requirements cannot be overemphasized: the non-prescriptive licensing regime allows flexibility but can be subject to interpretation and arbitrariness on both sides because there are no detailed "rules". Definition of licensing requirements forces agreement in areas where there are no recipes.
The vehicle for documenting regulatory requirements was the Licensing Basis Document (LBD). It contained a list of all the regulatory documents that would be applied, as well as national and international codes and standards. While logically agreement on the LBD should precede detailed discussions on design and safety analysis, in practice here (and later), a draft LBD was issued early on, but was not formally accepted until all its ramifications were understood, much later in the process and after a lot of detailed design decisions had been made. While not academically pure, this process was pragmatic and necessary.

Unfortunately there was no project commitment to Point Lepreau 2, for reasons unrelated to licensing risk, and the up-front licensing was not completed.

4.2. SES-10

SES-10 was a small 10 MW(th) pool reactor designed by AECL for urban district hot-water heating. It had extraordinary safety to meet extraordinary siting and operating requirements [11-12]. Safety characteristics included:

- negative reactivity coefficients of fuel temperature, coolant temperature, and coolant void[4];
- limited amounts of excess reactivity available to the reactor control system;
- natural convective heat removal with low flow velocities at all operating powers;
- a large passive heat sink (350,000 litre pool) for long-term emergency heat removal;
- low fuel ratings, resulting in negligible free fission products in the fuel;
- double-walled pool to prevent loss of water;
- a confinement barrier encompassing the pool structure and enclosing the top of the pool;
- very slow rates of reactivity control (a few mk/hour) to ensure accidents were slow;
- dual diverse shutdown systems, one active and one passive (thermally activated).

The siting and operating requirements included location in an urban environment with the reactor building boundary forming the exclusion area; and the ability to operate without a licensed nuclear operator in attendance. The reactor would be remotely monitored and could be remotely shut down; a local attendant could shut the reactor down in case of e.g., fire in the building or loss of communication with the offsite operator, but could not restart it. Once shut down, the reactor needed no "engineered" systems to remain safe.

Since the intent was to produce many of these reactors, it was not economic to licence each reactor separately. Moreover there were no relevant regulatory requirements for such small, safe reactors in Canada (the SLOWPOKE research reactor was the closest). Nor was there experience elsewhere, for that matter. A generic "up-front" licensing process was undertaken with the CNSC so that a "type-licence" could be obtained before commercial commitment. As before, the Licensing Basis Document served as the focus for the debate. Many very difficult issues were raised, and solved. For example, the CNSC set a requirement very early on that sudden removal of a control rod (by unspecified means, since there were no driving pressures and the rods were inaccessible during operation) must be within the design basis. Since another requirement was prevention of any fuel failure in a design basis accident, this led to a redesign of the reactor core, so that the worth of each control rod could be reduced to about 5 mk. The confinement concept, coupled with external event protection, resulted from the need to *have* a "containment" together with the lack of a design basis accident which would *require* a pressure-containment. Almost a year was spent on the operator model until the CNSC was satisfied that in principle it was workable.

Although the difficult concepts were worked out in the licensing process, the reactor was never commercialised due to the falling price of natural gas at the time.

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[4] The sign of a reactivity coefficient by itself is not an indicator of safety. For this urban heating reactor, however, AECL wished to ensure that to the extent practical, increases in reactivity were slow and self-limiting. The ultimate expression of this approach was achieved in the 20 kW(th) SLOWPOKE research reactor, where it was possible because of the low power density; as a result SLOWPOKE was, and is, licensed for unattended operation in cities.

4.3. CANDU 9

CANDU 9 is an evolutionary plant of capacity in excess of 900 MWe. It is based on the Darlington reactor core in a single-unit containment. It evolved from Darlington in much the same way as the single-unit CANDU 6 evolved from Pickering A. Since the market for CANDU 9 was initially outside Canada (specifically the possibility of Korea), AECL was faced with the challenge of providing to overseas purchasers assurance of licensability in the country of origin [13].

From the beginning, a number of fundamental requirements were set for safety and licensing, requirements that have moved CANDU 9 toward an internationally licensable product [14].

- The design had to be licensable in Canada. There were two reasons for this. First, it would make the product attractive to Canadian utilities when they began to build nuclear generating stations again. Second, although CANDU 9 is an evolutionary design, it was felt that utilities would still want an independent assurance of licensability, particularly focused on any changes, even if these changes were improvements. Thus the CNSC was asked to perform a formal licensability review to ensure that there were no “fundamental barriers” to licensing the CANDU 9 design in Canada. This conclusion would form the basis for the regulatory review done when a CANDU 9 was ordered, whether in Canada or overseas, and would assure the purchaser that the risk of significant design changes due to licensing was small.

- The CNSC recognized and supported the concept of “up-front licensing” as described above. The CANDU 9 assessment was the most extensive application of it to date.

- The design had to meet licensing requirements in the country which would eventually operate the plant. A Licensing Basis Document was written explicitly incorporating the requirements of both CNSC and KINS (the Korean agency responsible for regulatory assessment of the design on behalf of the Ministry of Science and Technology – MOST). KINS is an experienced regulator that had licensed both LWRs and CANDUs. Indeed, Korean licensing requirements had the biggest impact on containment design. CANDU 9 had to accommodate Korean siting practice and demonstrate a small Exclusion Area Boundary (EAB) of less than 500m.

- The design had to meet international licensing requirements, as embodied by the International Atomic Energy Agency (IAEA). There is, of course, no formal international licensing agency. However IAEA Standards and Guides are accepted as a starting point by increasingly more countries, and, when written into a bid specification, become *de facto* licensing requirements for the project. In China, IAEA guides have been used extensively to formulate the HAF guides. IAEA requirements tend not to be particular to a specific design, although they do reflect LWR practice to some extent.

- The design had to meet utility requirements for a modern evolutionary plant. In the U.S., EPRI has published a summary of utility requirements; in Korea, utilities have done likewise, using a similar framework. Although the specific requirements are tied to LWR designs, the general requirements are applicable to all water-cooled reactors, and were incorporated into CANDU 9.

- The design had to possess enhanced safety, especially in the area of severe accidents. Both the NRC and the IAEA had defined numerical targets for the frequency of core melt and large releases, and these were adopted for CANDU 9. Severe accidents were a particular focus of the LBD and subsequent requirements documents, and the high level requirements on the summed frequency of severe core damage were based on probabilistic goals:
    - 10^{-4}/year for 'moderator as a heat sink'[5]
    - 10^{-5}/year for severe core damage
    - 10^{-6}/year for large release.

To demonstrate licensability in Canada, and to assure overseas customers that the design had received independent regulatory review in the country of origin, the Basic Engineering Program included an extensive two year formal review [15] by the CNSC. Documentation submitted for this licensing review included the Licensing Basis, safety requirements, and safety analyses necessary to demonstrate compliance with regulations as well as to assess system design and performance. The first submissions were the Technical Description and the Licensing Basis Document (LBD). The LBD included not only requirements for licensability in Canada but also in the international market. The LBD, once accepted by the CNSC, provided both guidance to a foreign regulatory authority on how licensability in Canada is implemented, and also AECL's interpretation of additional requirements from the foreign authority. These two submissions were followed by more detailed design requirements, design methods (e.g., for safety critical software), safety analyses, probabilistic safety analysis, and other programme documents such as quality assurance, decommissioning, safeguards, and security requirements. In selected cases, CNSC inspected details of the design implementation. In total, over 200 formal documents were submitted. CNSC review of the detailed submissions, while comprehensive, focused particularly on:

- new or unique features in the CANDU 9 design,
- new or revised CNSC Regulatory or Consultative documents,
- Generic Action Items applying to all CANDU plants,
- known operational safety issues,
- importance to reactor safety.

Midway through the review, the CNSC staff identified thirteen key issues requiring a more detailed assessment. Intensive discussion took place for almost a year on these issues, resulting in many further submissions and analyses by AECL, and in some cases design changes, so that the issues could be closed at the end of the licensing review. At the end of the two years, CNSC issued a final detailed report summarizing the disposition of all issues raised, and any further commitments made by AECL. The summary of this report stated that:

> "[CNSC] staff conclude that there are no fundamental barriers to CANDU 9 licensability in Canada."

[5] This is a severe accident without fuel melting in CANDU, because of the presence of the moderator around the outside of the fuel channels.

This statement resulted from the review of the information provided to the CNSC, and was based on three general conclusions: that the CANDU 9 design complied, or could be made to comply with licensing requirements in effect, in Canada, on January 1, 1995; that the proposals to address CNSC Generic Action Items on the CANDU 9 design were acceptable; and that the major issues identified during the course of the licensing review had been adequately addressed.

The scope of the CANDU 9 licensability review was chosen carefully. Unlike Point Lepreau 2, AECL did not seek the equivalent of an operating licence from the CNSC. The reason was three fold:

- Final licensability would be determined by the country in which the reactor would be sited. KINS was an experienced regulator and did not need CNSC to certify every detail of the design. What was more important was that CNSC perform a thorough exploration of any changes in design concept and give assurance that the design met *current* Canadian requirements, even where these had not been applied to operating plants in Canada.

- Approval of design details would make it very difficult to make even minor changes, without some process for re-opening the licensability conclusion. Thus it risked freezing the design in time.

- Pareto's law applies: the extra effort from both CNSC and AECL to get approval of all design details would be very large, and the benefit very small. Indeed, CNSC does not approve all design details for a plant in Canada – instead (consistent with the Canadian safety philosophy outlined above – "proponent propose, regulator dispose") they audit selectively and deeply in areas of novelty or particular concern.

Nevertheless, CNSC identified a number of lesser items that were not sufficiently important to affect licensability, but which had to be addressed in the detailed design. AECL is presently addressing those items that affect the generic features of the CANDU 9 design and analysis, and a good number of them have been closed with CNSC acceptance of AECL's response. Items that depend on-site specific conditions and plant-specific customer requirements will be addressed once the project is committed [16].

5. LOOKING AHEAD

The CANDU 9 experience sets the pattern for other designs for which AECL wishes regulatory endorsement.

CANDU 6 plants undergo continuous improvement, so that the CANDU 6 plant offered today differs incrementally from those already operating. Moreover Canadian and international regulatory requirements continue to change. Thus AECL has asked CNSC to judge the acceptability of *changes* to CANDU 6 relative to what would be acceptable in Canada today. This is a much-reduced scope relative to CANDU 9, reflecting the fact that there are CANDU 6s operating worldwide.

AECL is also developing a Next-Generation CANDU (the "NG CANDU"), an evolutionary design but also a more significant departure from the current product line [17]. Designed to compete in the future open market place (and specifically therefore with natural gas), it retains all the CANDU 'basic' design features (horizontal pressure tubes, heavy-water moderator) but uses a light-water coolant operating at higher pressures and temperatures, and slightly enriched fuel. Whether the first customer is Canadian or not, the plant must be licensable in Canada and the licensing ground-rules must be understood. Up-front licensing will no doubt be pursued as the design is sufficiently developed.

6. LESSONS LEARNED FOR THE FUTURE

The experience in Canada leads to some key conclusions on the role and nature of risk reduction through successful up-front licensing. These lessons may or may not apply to other jurisdictions, because of the differences in history, licensing philosophy, and culture. They are:

- Up-front licensing is essential to reduce the risk of licensing-related delays once a project has been committed. It requires a significant investment in time and effort from both the designer and the regulator.
- The most effective scope for up-front licensing is for the regulator to thoroughly assess novel concepts, test the design against changed domestic requirements, and follow-up on known difficult areas; and for the designer to ensure foreign requirements are incorporated. It is of great benefit for the potential customer to be part of this process, to explain the requirements of his regulator and to ensure they are incorporated. There is little benefit in certifying the design in detail – customers really need assurance only for large issues; the effort to review each detail is disproportionately large; and the ability to make improvements is hampered.
- Although it would be satisfying to have legally-binding certification, in the end there can be no legal obligation on the regulator, who would not accept such constraint in the face, for example, of new information from R&D or world experience. The agreement is pursued on the basis of good faith that the regulator will not make arbitrary decisions and that the designer will carry out his commitments. One cannot decree good faith but one can create it.
- In almost all circumstances, issues will arise that are beyond the current 'rules', however expressed, sometimes because the 'rules' have not kept pace with international or domestic developments; or because the design poses novel challenges. The Canadian licensing flexibility has worked well here, so that rather than rushing to create new rules, one reaches a sensible conclusion. The conclusion becomes precedent; precedent becomes practice; practice becomes a regulatory guide.
- PSA, and its companion, cost-benefit analysis, will play an increasingly important role in up-front licensing, as a way of casting decisions in an objective framework. Not all decisions will be made this way, but a lot will be influenced by these tools.
- International licensability would be a robust form of up-front licensing. Currently IAEA Safety Guides provide a lowest-common-denominator for most regulators. The combination of meeting the requirements in the country of origin, of a mature purchaser, and of the IAEA constitutes *de facto* international licensability.
- AECL sees continued advantages to all in pursuing such approach in order to allow acceptable risks both for the regulator and the customer, and to enable innovations and improvements in safety to evolve naturally and without undue legal, design or regulatory constraints.

- Canada started its nuclear safety philosophy by comparing the risks of various energy sources. In a competitive energy market, it is inappropriate to place undue safety requirements on one technology compared to the others. In the longer term, regulatory rationalization across all means of energy production would be of net benefit to society.

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Annex 16

Trends and needs in regulatory approaches for future reactors

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Abstract. For nuclear power to be the future alternative of choice for electric generation capacity, there are two essential elements: (1) cost competitiveness, and (2) an acceptable level of safety. To meet these perhaps conflicting elements, there will need to be significant risk-informed modifications to the regulatory approaches for the licensing and oversight of reactors. This paper discusses some of the trends in the U.S. in this direction and identifies what the author believes will be technical and policy issues that stand in the way.

1. INTRODUCTION

As a result of concerns over the effects of air pollution and the need for an assured energy supply, nuclear power is beginning to receive increased attention as an attractive alternative for new electric power generation capacity. Whether or not new nuclear plants prove to be the alternative of choice will depend, I believe, to a large extent on two elements: (1) the cost competitiveness of nuclear over the alternative choices (e.g. coal, oil, and natural gas), and (2) an increased level of safety that is transparent and fully embraced by the general public. The level of safety to satisfy the second element has not been established. I believe, however, that new designs will have to be such that evacuation of the surrounding population will not be required to meet risk acceptance criteria. To satisfy the "transparency" requirement, the actual risk status of each plant will have to be quantified in a way that can be easily communicated to the general public and must satisfy broadly accepted risk limits on an individual plant basis.

The use of such risk acceptance criteria has not been a major explicit part of the current licensing and regulatory process in the U.S. Instead, the regulatory structure has been described as being "deterministic" in nature. It basically consists of a set of deterministic requirements on the design and operation that include a defined set of "design basis accidents" (DBAs) which must be analyzed and designed against to satisfy certain "figures-of-merit" (e.g. peak fuel clad temperature, hydrogen production, dose limits). The set of General Design Criteria (GDC of Appendix A of 10CFR50 spell out defense-in-depth constraints on how the plants must be designed to meet the figures-of-merit acceptance criteria for the DBAs. Indeed, throughout the whole of the regulatory system, the defense-in-depth[2] philosophy is followed. The overall criterion for license acceptability is that the plant must meet the "adequate protection" standard.[3] It is admitted that this regulatory system has been cleverly devised and improved upon with years of experience and it has resulted in a level of safety for the U.S. plants that is considered by most to be acceptable. This regulatory system can also be said to possess the following characteristics:

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[1] Opinions expressed in this paper are the author's own and do not necessarily represent the positions of either the U.S. Nuclear Regulatory Commission or the Advisory Committee on Reactor Safeguards (ACRS).

[2] Defense-in-Depth as currently defined by the USNRC is an element of NRC's safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The concept includes consideration that safety not be **wholly** dependent on any single element of the design, construction, maintenance or operation of a nuclear facility.

[3] "Adequate protection" of the public health and safety is said to be provided if the plant's design meets all the regulatory requirements.

(1) The system may be overly conservative and may have placed significant unnecessary burden on some licensees.

(2) In places, the system lacks coherence such that the efficiency of the regulatory process suffers.

(3) Because of the complexity and relative lack of coherence of the system and the somewhat legalistic and bureaucratic circularity associated with the concept of "adequate protection", it is not apparent to the general public that the level of safety is acceptable nor do they have a firm understanding as to what level of safety has actually been achieved.

(4) The plant designs that have resulted from meeting the "adequate protection" standard have a relatively broad distribution of risk statuses and of their defense-in-depth balance.

(5) The regulations are primarily focused on light water reactors and may not be entirely appropriate for other design concepts.

In an economically deregulated environment, there is likely to be tension between the dual needs of cost competitiveness and acceptable safety. To deal with this tension and to assess licensability of new reactor concepts, I believe new regulatory approaches to licensing and changes to the licensing basis will be required. The U.S. has already embarked on a "risk-informed" regulatory approach that has some of the elements necessary for this.

2. REGULATORY TRENDS IN THE US

2.1. Risk-informed changes to the licensing basis

The U.S. Nuclear Regulatory Commission (NRC) policy statement on probabilistic risk assessment (PRA)[1] encouraged greater use of PRA to improve safety decision-making and to improve regulatory efficiency. As one response to this policy statement, the NRC staff developed an alternative to the standard exemption process for licensees to use to support requests for changes to the plant's licensing basis. This alternative approach, described in Regulatory Guide (RG) 1.174 [2], has been called "the risk-informed approach".

In this approach, acceptability of licensing basis changes that may result in increasing the risk status of a plant is partially judged on the basis of both the plant's current risk status and the amount of the increase in risk that would result from the change. The risk metrics used in this risk informed approach are core damage frequency (CDF) and large early release frequency (LERF)[4]. In addition to the risk considerations, RG 1.174 places other constraints on the process. These involve maintaining acceptable margins, maintaining defense-in-depth, monitoring of the impact of the change, and meeting all other regulatory requirements.

With some modification and additional qualifications discussed below, this approach could form the basis for regulatory decision making related to cost-cutting measures to make operating plants more cost competitive while maintaining an acceptable level of safety.

[4] LERF is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. It can be thought of as a surrogate for NRC's early fatality quantitative health objective of the Safety Goals.

2.2. Risk-informing the US regulations

In 1998, the NRC formally defined risk-informed regulation as "an approach to regulatory decision making that uses risk insights as *well as traditional considerations* to focus regulatory and licensee attention on design and operational issues commensurate with their importance to health and safety"[3]. The NRC staff is currently involved in activities aimed at determining how best to provide a risk-informed alternative to be body of regulations as comprised by 10CFR50. Such an alternative could be used either by new plants for obtaining a license or by currently operating plants if it is to their advantage to voluntarily change en toto from the current system to the alternative one.

Although this activity of risk-informing the body of regulations is still a work in progress, the staff has discussed a framework for doing this [4]. A representation of this framework is given by Figure 1 taken from Reference 4. As can be seen in this figure, this is a top-down approach using "adequate protection" as the top level goal and utilizing "defense-in-depth" as the overriding philosophical approach. The definition of adequate protection in this context is still intended to be as an un-quantified compliance with the regulations, and quantitative goals would not generally appear in specific regulations. Defense in depth at the high level is still intended to be a "balance" between prevention and mitigation by application of four strategies:

1. limit the frequency of accident initiating events,
2. limit the probability of core damage given accident initiation,
3. limit radio-nuclide releases during core damage accidents, and
4. limit public health effects due to core damage accidents.

The focus here is on controlling those accidents that could otherwise result in large public exposures. Presumably, other regulations intended to protect nuclear plant workers and the public during routine operations and to provide physical protection against sabotage threats will be maintained as they currently exist.

The intent of the effort is to develop requirements that retain the deterministic characteristics in such a way that compliance will still be the measure of adequate protection. Quantitative limits will not generally appear in the regulations. Nevertheless, in developing the "deterministic-like" requirements, the staff will rely heavily on the Commission's Safety Goals with the understanding that " ... replacing existing regulations with the quantitative health objectives (QHO) would ... not assure defense-in-depth against limitations and uncertainties inherent in PRA." For guidance, the staff intends to utilize subsidiary (or surrogate) goals that are consistent with the QHOs. One concept for this that has been considered is shown in Figure 2 also taken from Reference 4.

The remarkable feature of Figure 2 is the focus on limiting the conditional core damage probability (CCDP) and the conditional early containment failure probability (CECFP) for ranges of initiating event frequencies that are also goals for these initiators. The intent is that the risk-informed deterministic regulations will be so cleverly crafted that compliance with these should provide a *reasonable expectation* that the quantitative goals will be met.

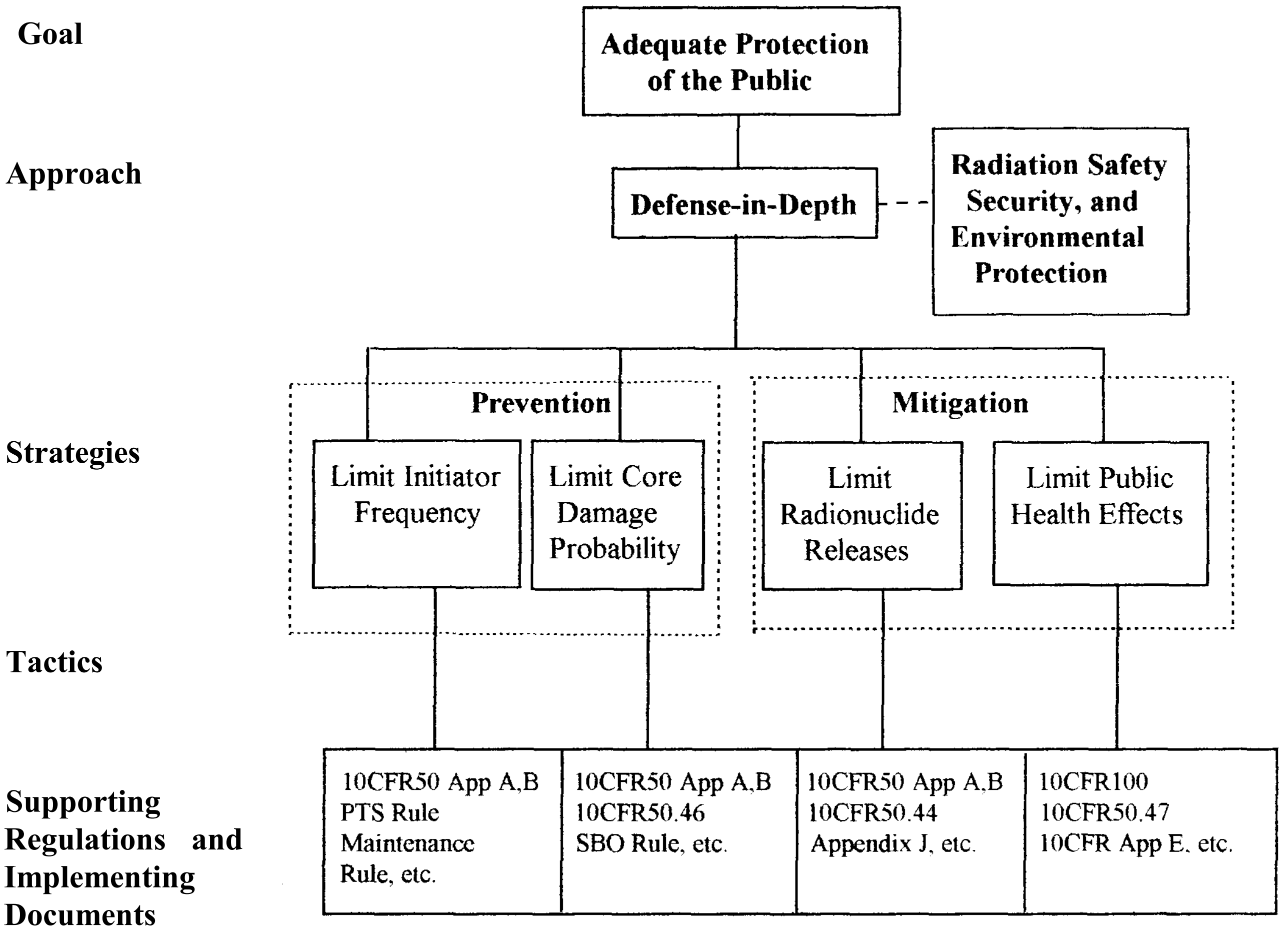

Figure 1. Framework for risk-informed regulation.

3. REGULATORY NEEDS FOR RISK-INFORMED REGULATIONS

Some of the concepts and principles in both RG 1.174 and the above risk-informed regulatory approach can go a long way toward providing the framework that will permit economic optimization while maintaining acceptable safety. The risk informed approach discussed above actually differs little from the current deterministic system - it just provides a better focus. Thus, in my opinion, it will not resolve the issue of circularity associated with the adequate protection concept nor will it provide the necessary assurance to the general public that the regulations do, indeed, result in acceptable risk.

3.1. PRA Needs

For risk-informed concepts to be properly implemented, much has yet to be done. In the first place, such a risk-informed regulatory system will be highly reliant on PRAs and their bottom-line results. **Before such major reliance is tenable, there must be international consensus on what constitutes an acceptable quality PRA and how to assess that quality. In addition, a number of technical improvements to PRAs are needed that include:**

Quantitative Health Objectives (QHOs)
Early Fatality Safety Goal
$\leq 5 \times 10^{-7}$/year
Latent Cancer Safety Goal
$\leq 2 \times 10^{-6}$/year

(1) Prevention-Mitigation Assessment, Consider the Strategies In Pairs	**Prevent**	**Mitigate**
	Core Damage Frequency $\leq 10^{-4}$/year	**Conditional Early Fatality Probability $\leq 10^{-2}$**

(2) Initiator-Defense Assessment, Consider the Strategies Individually (Preferred)	**Limit the Frequency of accident Initiating Events (Initiators)**	**Limit the Prob. of Core Damage Given Accident Initiation**	**Limit Radionuclide Released During Core Damage Accidents**	**Limit Public Health Effects Due to Core Damage Accidents**
	Initiator Frequency	**Conditional Core Damage Probability**	**Conditional Early Containment Failure Probability**	**Conditional Early Fatality Probability**
Anticipated Initiators	$\leq$ 1/year	$\leq 10^{-4}$	$\leq$ 0.1	$\leq$ 0.1
Infrequent Initiators	$\leq 10^{-2}$/year	$\leq 10^{-2}$	$\leq$ 0.1	$\leq$ 0.1
Rare Initiators	$\leq 10^{-5}$/year	$\leq$ 1	$\leq$ 1	$\leq$ 0.1

Note: The product across each row gives $\leq 10^{-6}$/year.

Figure 2. Quantitative goals for risk-informing regulatory requirements.

- The ability to assess the total risk including shutdown and low power, fires, seismic, and safeguards.
- A methodology for including safety culture, economic and organizational factors as well as better treatment of human factors.
- A methodology to better include aging effects.
- The ability to incorporate a full uncertainty assessment (including both aleatory and epistemic uncertainty).

In addition to these PRA needs there **are a number of policy and technical issues that need better resolution to have a coherent risk-informed regulatory system.**

3.2. Policy and Technical Issues

3.2.1. Risk acceptance criteria

It is a mistake for the regulatory system to still insist on making the legalistic concept of adequate protection as its top level acceptance criterion. That this condition persists is, I believe, one of the major contributors to the public's lack of understanding of the level of safety achieved by the various plants. In my opinion, it leads to an impression of legalistic double-talk that hurts the agency's credibility. Since it is the overriding mission of nuclear regulatory agencies to assure no undue *risk* to the health and safety of the public, **it would make abundant sense that there be quantified high-level risk acceptance criteria that each plant site be required** to meet. This should be a separate requirement in the regulations along with any traditional deterministic requirements. It is beyond my ken why this simple concept has been so strongly resisted.

The quantitative guidance in the NRC risk-informed approach does not appear to go far enough in development a complete set of risk acceptance criteria. Core damage frequency (CDF) and large early release frequency (LERF) are not complete regulatory objectives. The sets of deterministic regulations that currently exists also deal with latent effects to a larger population, land contamination, food interdiction (all of which are associated partially with late containment failure) as well as injuries and worker exposure. **In a risk-informed system, these must be preserved in some way.**

The choice of initiating event frequency intervals appears to be too subjective and too broad (a range of two orders of magnitude?). In principle, with modern PRAs, **there is no reason acceptance criteria cannot be placed on the entire spectrum of frequencies** much as noted by the IAEA in the use of frequency-consequence (FC) curves [5]. In addition, there seems to be little consideration of uncertainties associated with the quantification among sequences. **It would, for example, make good sense to render those sequences with high uncertainty to much lower percentage contribution to risk compared to those sequences with less uncertainly.** This is a kind of "rationalist" defense-in-depth concept.

There is a need for international consensus on the technical basis to be used to develop consistent risk acceptance criteria. There should be an underlying basis of risk/benefit for these and they should represent societal values in the sense that the limits represent what risk the affected public is willing to accept at some assurance level given the benefit of nuclear power to that public. It is entirely rational that such criteria may differ among countries with different circumstances. The full distribution for each risk metric needs determination so as to place confidence values on the level achieved. **If acceptance criteria do indeed include allocation among sequences and events, there needs to be incorporated some concept of optimization from the viewpoint of uncertainty contribution.**

3.2.2. Defense-in-Depth (DID)

The definition of defense-in-depth used by NRC (see the footnote on page 1) has been termed by ACRS in one of its recent reports [6] as the "structuralist" view. This view appears to be largely preserved in the agency's proposed approach for risk-informing 10CFR50. In Reference 6, ACRS articulated an alternative view of defense-in-depth called the "rationalist" view. This view asserts that defense-in-depth is the aggregate of provisions made to **compensate** for uncertainty and incompleteness in our knowledge of accident initiation and progression - i.e. the uncertainty associated with PRA results. **In order to avoid arbitrary appeals to DID and prevent it from becoming a hindrance to the optimization of nuclear plant designs, this rationalist view needs to be more widely accepted but also further developed and better articulated. There needs to be some technical basis for establishing the necessity and sufficiency of defense-in-depth measures in a quantified manner**. If one closely examines the current structuralist view as defined on page 1, it is seen that it fundamentally amounts to an un-quantified allocation of risk among sequences and various elements of those sequences. In effect, this is also what Figure 2, in the risk-informed approach attempts to achieve. The structuralist and the rationalist views are coming close together in this respect. **These could be rationalized**

and unified if the regulatory requirements were made to be limits on the frequency associated with any given consequence (dose) at appropriate confidence levels which could vary as the consequence increases (i.e. an expansion of the FC concept). This, in essence, would be a built-in and quantifiable defense-in-depth concept that would be independent of reactor type. It would have to be recognized that some regulatory objectives are not well suited for quantification by PRA (e.g. QA, monitoring, inspection and testing, sabotage, etc.). There would have to be retention of deterministic requirements for these. There needs to be recognition that, for some events, complete reliance should not be placed on PRA bottom-line results. These would be for those events that are of potentially high consequences and for which both the frequency and the consequences have high aleatory (knowledge-based) uncertainty. One approach for implementing a concept similar to this can be found in a paper presented at the recent Nuclear Safety Research Conference [7].

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Annex 17

A completely new design and regulatory process — A risk-based approach for new nuclear power plants

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Abstract. In the de-regulated electric power market place that is developing in the USA, competition from alternative electric power sources has provided significant downward pressure on the costs of new construction projects. Studies by the Electric Power Research Institute have shown that, in the USA, the capital cost of new nuclear plants must be decreased by at least 35% to 40% relative to the cost of Advanced Light Water Reactors designed in the early 1990s in order to be competitive with capital costs of gas-fired electric power plants. The underlying reasons for the high capital costs estimated for some nuclear plants are (1) long construction times, (2) the high level of "defense-in-depth" or safety margin, included throughout the design and licensing process, and (3) the use of out-dated design methods and information. Probabilistic Safety Assessments are being used to develop a more accurate assessment of real plant risk and to provide relief if it can be demonstrated that plant equipment is not providing a significant contribution to plant safety. Westinghouse addressed some of these cost drivers in the development of the AP-600 passive plant design. However, because of relatively inexpensive natural gas plant alternative, we need to reduce the costs even further. Therefore, the AP-600 design is now being up-rated to a 1000 MWe design, AP-1000. The development of AP1000 is described in another paper being presented at this meeting. Westinghouse is also managing a project, sponsored by the US Department of Energy, which is aimed at developing an all-new "risk-based" approach to design and regulation. Methodologies being developed use risk-based information to the extent practical and "defense-in-depth" only when necessary to address uncertainties in models and equipment performance. Early results, summarized in this paper, include (1) the initial framework for a new design and regulatory process and (2) a sample design analysis which shows that the Emergency Core Cooling System can be eliminated with its safety function performed by the normal makeup water system.

1. INTRODUCTION

Risk information has always been factored into the design of nuclear power plants. In the early days of the industry (i.e., ~1960s) risk was primarily addressed in a qualitative manner. For example, certain material characteristics, equipment reliability, and plant performance under accident conditions were not well known. To resolve these and other uncertainties, margin was added during the design of structures, systems and components. Examples are the use of load factors and combination of worst-case loads in the seismic design of structures, use of redundant trains of safety systems, and development of very conservative deterministic safety analysis methods. Now we have new technology (e.g., materials, experimental data) and new design methods such that previous design and regulatory assumptions can be re-assessed in the light of Probabilistic Safety Assessments (PSAs) of the whole plant.

Westinghouse initiated its advanced, AP600 passive plant design program to address the worldwide nuclear industry requirements such as increased safety, increased reliability, and improved economics using proven passive technology in combination with detailed risk assessments. In addition, the resulting design would have to meet the basic licensing requirements of countries around the world as well as those of the United States. The design strategy for Westinghouse passive plants was based on meeting globally recognized customer requirements such as the U.S. Advanced Light Water Reactor Utility Requirements (ALWR URD). Designers also incorporated the extensive lessons learned from operating reactors in the U.S., Asia, and Europe. A concerted effort was made to simplify systems and components to facilitate construction, operation, and maintenance. State-of-the-art, proven technology was used to solidify confidence in the high performance expectations of the passive plant designs. Passive plant safety systems performance relies on the natural forces of gravity, natural circulation, and evaporation to shutdown and cool down the plant in the unlikely event of an accident. Passive systems also contribute to improved plant economy through simplification, while still meeting regulatory and public acceptance criteria. Following this strategy, Westinghouse developed an advanced passive pressurized water reactor (PWR) plant. The AP600 is a two-loop, 600-MWe plant designed in collaboration with the U.S. Department of Energy (DOE), Electric Power Research Institute (EPRI), the Advanced Reactor Corporation, and 22 international partners. To improve plant economics, the AP1000, a two-loop 1000-MWe configuration, is being developed using the AP600 design and licensing bases to the maximum extent possible.

Detailed risk assessments were performed using state-of-the-art methods that were summarized in the ALWR URD in the early 1990s. The resulting AP600 PSA was the major analytical tool for assessing risk improvement capability of plant structures, systems and components and for assessing the overall level of safety relative to URD goals for core damage frequency and large offsite release frequency. Whereas the NRC's criterion, or limit, on core damage frequency for internal events is 1xE-4 events/year, whereas the core damage frequency for currently operating plants is in the range of 5E-5 events/year, and whereas the core damage criterion for ALWRs is 1E-5 events/year, the core damage frequency (internal events) for AP600 is 3E-7 events/year[1]. Therefore. the AP600 design has achieved an improvement in safety by a factor of 10 to 100 over current and evolutionary-generation plants.

2. A RISK-BASED APPROACH TO NUCLEAR PLANT DESIGN AND REGULATION FOR DEVELOPMENT OF NEW NUCLEAR POWER PLANT DESIGNS

The development of the AP600 and AP1000 designs has achieved significant cost reductions. However, these design efforts have been carried out in the context of today's regulations, making changes to regulatory guidance as needed during the regulatory review process. Through the Nuclear Energy Research Initiative (NERI), the U. S. Department of Energy is sponsoring a series of projects aimed at further reducing the costs for a new generation of nuclear power plants. The intent of these projects is to develop new reactor designs and to take a fresh look at the design and regulatory processes needed for efficient development and implementation of those designs.

Three NERI projects with the same goal of cost reduction are the "Risk-Informed Assessment of Regulatory and Design Requirements for Future Nuclear Power Plants"[2], the "Smart Nuclear Power Plant Program"[3], and the "Design, Procure, Construct, Install and Test" (DPCIT) Program[4].

2.1. Smart project overview

The goal of this program is to design, develop, and evaluate the methods for implementing smart equipment and predictive maintenance technology. In this program, "smart" equipment means components and systems that are instrumented and monitored to detect incipient failures in order to improve their reliability. The resulting smart equipment methods will be combined with a more risk-informed regulatory approach to allow plant designers to simplify designs without compromising overall reliability and safety. This concept will allow designers to address reliability at the component and system level while reducing dependence on costly practices such as redundancy and diversity of safety systems.

This program began with a system evaluation and prioritization study that identified and prioritized nuclear plant equipment which would most likely benefit from the addition of "smart" features (e.g., sensors, data processing, and man-machine interface devices). An optimum equipment health-monitoring system is being developed for a selected component (i.e., a normally operating horizontal centrifugal pump). A "virtual machine" methodology to simulate equipment behavior for evaluation of the overall benefits to system performance will be developed in the next phase of this project.

Also, methodologies will be developed for consolidating and presenting the data obtained from "smart" equipment to ensure that the health of the “smart” plant is readily understandable and is consistent with existing Man-Machine Interface (MMI) methods. A survey has been conducted to determine how smart equipment information is presented to users in other industrial applications; results are now being evaluated and applicable characteristics will be adopted for this project.

The final task in this program is the expansion of the concept of smart components to system and plant-wide levels. While it is beneficial to perform health monitoring on individual pieces of equipment, the ultimate goal is to develop methodologies to combine health-monitoring information into a plant-wide system.

2.2. DPCIT project overview

Reduction of the complete design-testing cycle for new nuclear plants is the goal of this project. The key objectives are (1) leveraging Information Technology, (2) determining the impact on schedule reduction of long-lead-time items and possible remedies, (3) incorporation of insights from manufacturing, (4) linking 3D Computer Assisted Design to Project Management tools, (5) applying conceptual ideas such as modular construction, (6) examining potential the critical path and determining how to eliminate interfaces that cause substantial rework, (7) adopting an electronic commerce business model in which suppliers and the design/manufacturing organization are not just linked, but also in which work is performed in parallel paths, and (8) determining the applicability of finite element analysis to identify potential improvements in nuclear containment structures that would allow significant reductions in capital cost.

This program will achieve its goals by questioning “how” and “why” work is performed. A DPCIT cycle will be adopted as the point of reference in the investigations. In the final analysis, any proposed improvements must point towards meaningful reductions in the length of time and total cost of the DPCIT cycle. This method of accounting forces all costs and time to be rolled up for impact on the project, thus avoiding the problems of sub-optimization of individual components at the expense of the overall goal. The merger of the potential improvements in the DPCIT cycle will be expressed in a series of models to describe how the improvements can be implemented for the next generation plant.

2.3. Risk-informed project

The primary objective of this project is the development of methods for a new, highly risk-informed - or “risk-based” - design and regulatory process. Past risk assessments were generally used to assess designs at the end of the design process, due to lack of comprehensive data, analytical models, and/or acceptance criteria. These risk assessments supported the historic deterministic design and regulatory process. Those deterministic methods resulted in design margin and features being added to resolve safety issues such as the large break loss-of-coolant accident and severe accident mitigation without detailed consideration of the impact on plant cost. The NRC and industry are now conducting a program to "risk-inform" the regulation of today's operating plants. This effort will hopefully result in regulatory relief when justified by PSA analysis and other current information. This approach to risk-informing the regulations starts from the existing design and regulatory process and justifies each specific change to those bases. The NERI Risk-Informed project team believes that the currently ongoing process may be appropriate for operating plants, but is not appropriate for the development of new reactor designs. Therefore, this project has embarked on a "clean sheet of paper" approach, which can be characterized as a complete re-generation of the design and regulatory process using risk-based methods to the maximum extent practical, including a re-assessment of all previous design and regulatory methods and assumptions.

2.3.1. Risk-informed project task summary and participants

This project includes two basic tasks which are summarized below. The task participants are also identified below.

2.3.1.1. Task 1: Development of risk-informed methodologies

Many of the regulatory requirements and industry standards that form the bases for designing the current generation of nuclear plant designs are based upon subjective, deterministic assumptions that were limited by the knowledge-base and engineering tools that were available at the time that those requirements and standards were created. The research effort proposed for this project is to develop a set of risk-informed methodologies that can be used by future plant designers to (1) systematically develop and/or utilize all of the regulatory requirements and industry standards that would impact the design of new nuclear plants and (2) systematically develop designs for a nuclear plant's SSC's, by applying those methodologies. This research effort will be complementary to the current industry/NRC efforts to apply risk-informed, performance-based regulation to selected issues that affect operation of existing nuclear plants. The methodologies developed in this research project will then be demonstrated, by applying them to a sample problem. The methodologies may then be revised to apply the lessons learned from this sample. Task 1 includes the following subtasks:

- Subtask 1.1: Identify applicable current regulatory requirements and industry standards;
- Subtask 1.2: Identify systems, structures, and components (SSCs) and their associated costs for a typical plant;
- Subtask 1.3: Develop methodology for developing risk-informed requirements and standards;
- Subtask 1.4: Develop methodology for designing highly risk-informed SSCs;
- Subtask 1.5: Identify high priority requirements, standards, and SSCs;
- Subtask 1.6: Apply methodologies to a sample SSC;
- Subtask 1.7: Evaluate regulatory processes and develop recommended improvements;
- Subtask 1.8: Coordinate activities with ongoing efforts of NEI, NRC, and industry.

2.3.1.2. Task 2: Strengthen the reliability database

To fully risk-inform the design bases for future nuclear plants, it is essential that the reliability database for the SSC's be complete. Current industry/NRC efforts to strengthen the reliability database are primarily focused upon issues that affect operation of the existing nuclear plants. The research effort proposed for this project will identify where strengthening of the risk assessment database is needed to support the design of new plants – including identification of the reliability information that will be needed to support introduction of new, advanced "smart" technologies. The research effort will also recommend programs for collecting the information that will be needed by future plant designers, to provide this information. Task 2 includes the following subtasks:

– Subtask 2.1: Identify current sources of reliability data for SSCs;
– Subtask 2.2: Identify weaknesses in sources;
– Subtask 2.3: Develop industry/government programs for correcting the weaknesses.

2.3.1.3. Task participants

The team for this project was selected to provide a wide range of technical capability and innovative ideas for developing a new design and regulatory process. The team comprises the following representatives from industry, national laboratories, and universities:

– Westinghouse, as the lead organization, provides overall coordination and project management. It also provides expertise on the design and analysis of systems for nuclear plants and the licensing of nuclear plants;
– Sandia National Laboratories (SNL) provides expertise in risk methodology development, especially as it affects structures, low power and shutdown operations, fire risk, and object oriented risk and reliability analysis methodology;
– Idaho National Engineering & Environmental Laboratory (INEEL) provides expertise in risk methodology development, risk analysis tool development, and data collection and assessment methodology development;
– Massachusetts Institute of Technology (MIT) provides expertise in structuring the approach to risk-based regulation, the selection of design basis events in a risk-based process, and the strategy for building the needed PRA database;
– North Carolina State University (NCSU) provides expertise in aging and structural analysis;
– Duke Engineering and Services (DE&S) provides expertise on the design and construction of systems and structures for nuclear plants and the evaluation of performance data;
– Egan & Associates, P.C. provides expertise in nuclear law, nuclear licensing and nuclear regulation.

2.3.2. The new risk-based design and regulatory process

The heart of the new risk-based design regulatory process is the development of methods by which PSAs can be used to remove excessive conservatism, simplify plant designs, lower their cost, and at the same time maintain a high level of safety. Methodologies being developed use risk-based information to the maximum extent practical and "defense-in-depth" only when necessary to address uncertainties in PSA models and equipment performance. Early results, summarized below, include (1) the initial framework for a new design and regulatory process and (2) a sample design analysis which shows that the Emergency Core

Cooling System can be eliminated with its safety function performed by the normal makeup water system, crediting a lower pipe break probability and using "smart" charging pumps.

A diagram of the risk-based design and regulatory process is shown in Figure 1. The major features of this process include:

- Retention of the current concepts of safety margin and adequate protection of the public health and safety. The method for meeting these goals, however, will be significantly revised;
- Establishment of probabilistic safety goals to demonstrate compliance with the adequate protection goal. Previously, the main method for assuring adequate protection included a set of deterministic criteria combined with judgments on safety margin and application of defense-in-depth. In the new regulatory framework, the use of probabilistic safety goals would be the primary means of assuring plant safety;

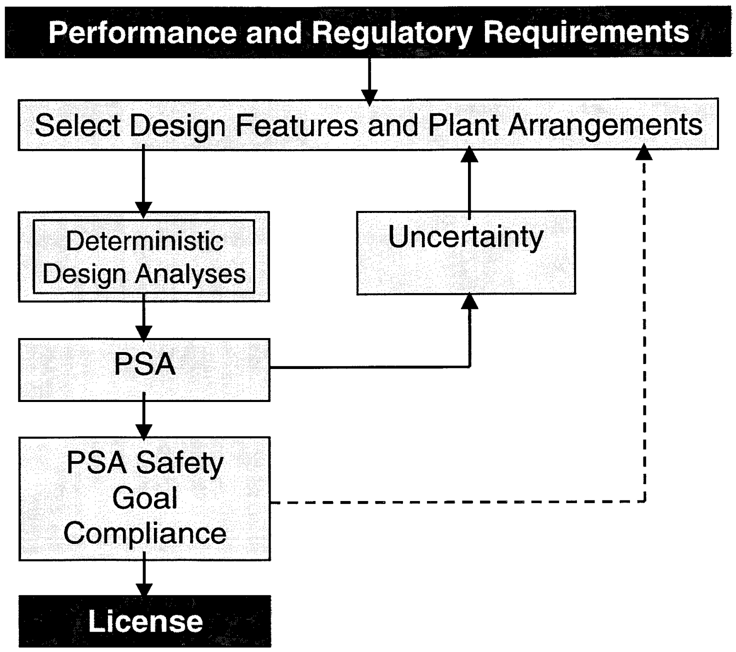

FIG. 1. Overview of the risk-based design and regulatory process.

- Specification of PSA methods to evaluate compliance with the safety goals. These methods would be consistent with those currently in place for ALWRs and operating plants, however, improved methods of addressing uncertainties would be implemented;
- Retention of the basic prevention and mitigation concepts when selecting specific probabilistic goals to be used in the design process. Regulatory policy may require the use of defense-in-depth in certain situations, regardless of what results are obtained from use of the above PSA methodology. For example, it may be appropriate to specify a core damage frequency goal and a large offsite radiological release goal to allocate risk between prevention and mitigation systems. Doing this would require the inclusion of a containment in the design of a plant, even if the PSA analysis demonstrated that the overall plant safety goal could be met using only core damage prevention features and systems. It is recognized, however, that such assumptions may be specific to a type of reactor design and that a truly generic process would address only those features common to all types of reactor designs;

- Use of PSA risk-based methods to resolve all uncertainties and margins to the maximum extent possible; that is, use of defense-in-depth only when uncertainties cannot be resolved with risk-based methods. It is realized that PSA methods and specifically the treatment of uncertainties is not perfect. Also, equipment performance under all potential conditions is not perfectly understood. If such uncertainties can be resolved in a cost-effective manner, new programs or research would be proposed. Otherwise, defense-in-depth and the inclusion of safety margin in the design would be used to resolve these uncertainties, but only when risk-based methods could not be used;
- Establishment of the corresponding regulatory criteria. It is envisioned that a new set of regulations and regulatory guidance would be developed consistent with the new, highly risk-informed design and regulatory process. These regulations and guidance would most likely be developed based on writing of new regulatory documents rather than revising existing documents.

2.3.3. Risk-informed design analysis

The key principles for development of a risk-based design are:

- Use the new risk-informed design and regulatory framework from the very beginning of the design process (i.e., use a completely new design approach);
- Do what is technically correct and justifiable and resist use of arbitrary conservatism and design margin;
- Evaluate all major assumptions, criteria, and safety margins, affecting the cost of a nuclear power plant;
- Maintain a level of plant safety at least equivalent to that required of today's ALWRs such as System 80+, KNGR, ABWR, and AP600.

The System 80+ PSA model was used for a sample problem analysis. A new approach to mitigation of a LOCA was investigated. This sample problem (1) relies on more recent data on pipe rupture probability to justify lower initiating event probabilities, (2) credits "smart" monitoring of the charging pumps to justify the high reliability needed for equipment performing safety functions, (3) credits leak-before-break technology to justify the removal of the large double ended pipe rupture from the plant design basis, and (4) evaluates the risk from large pipe ruptures even though they are not in the design basis. Figure 2 shows a safety injection system typical of an advanced pressurized water reactor. It is a four-train system dedicated to the safety injection function. Figure 3 shows an alternative, advanced design that eliminates most of the equipment associated with the dedicated safety injection system, but fulfills the safety injection function using the normally-operating, high-pressure charging pumps with variable-speed drives that provide a wide range of discharge flows. This design change results in an equipment cost savings of approximately $15 million to $20 million. There would also be substantive cost reductions for plant structures.

From Table I, it can be observed that the initiating event frequency is decreased by about one order of magnitude when the lower initiating event probabilities are used. The core damage frequency results show that the beneficial effects of the lower initiating event probabilities are offset by the effects of having less mitigation equipment in the design. A single train of the advanced design can adequately mitigate LOCAs for break sizes less than or equivalent to a double-ended rupture of a 10-inch diameter pipe. Therefore, since the larger pipe breaks are not included in the plant design basis, the single failure criterion need not be applied to the design of the advanced system.

TABLE I. LOCA CDF COMPARISON

Initiating Event	System 80+ ECCS		Surrogate Advanced Design	
	IE Freq (1) [Per Year]	CDF [Per Year]	IE Freq (2) [Per Year]	Quantified CDF [Per Year]
Large LOCA	6.97E-05	1.09E-07	5.00E-06	1.49E-07
Medium LOCA	1.49E-04	3.02E-07	8.92E-05	1.18E-07
Small LOCA	3.00E-03	1.97E-07	5.00E-04	4.44E-07
Total		6.08E-07		7.11E-07

1. EPRI Key Assumptions and Ground-rules Data (ALWR URD)
2. INEEL Data (NUREG/CR-5750)

The above sample problem results would be acceptable in a risk-informed regulatory framework since the total LOCA core damage frequency is essentially unchanged. Removal of the large pipe breaks from the plant design basis will likely lead to the need to resolve other regulatory issues such as qualification of non-safety, "smart" equipment (e.g., charging pumps) and increased review of other events which challenge fuel or reactor coolant pressure boundary integrity.

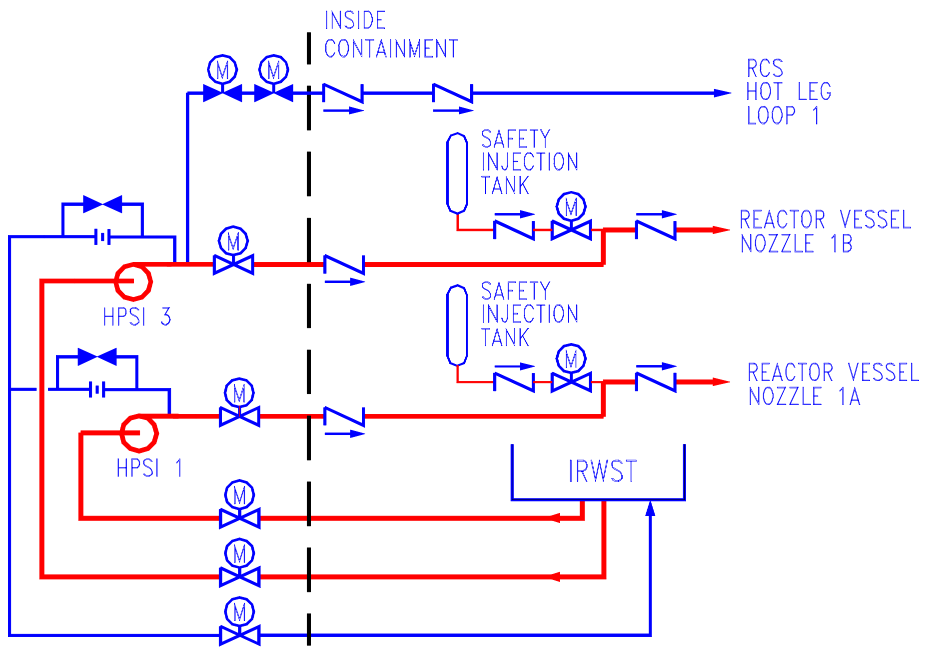

FIG. 2. System 80+ safety injection system (one of two divisions).

Further, a substantive development and implementation effort must be carried out. This effort will have to include the development of smart equipment (pumps and valves) and the development of new regulations and guidance. Issues to be addressed include the combination of safety and non-safety functions in a single system, expanded use of Leak-Before-Break technology, elimination of the single failure criterion, and determination of which "design basis" events should be used to assess the plant's design performance. It is believed, that a new risk-based process can lead to lower plant costs and a realistic assessment of plant safety only if the above issues, which have been used for the past several decades, are re-evaluated and replaced with today's technology and current analysis methods.

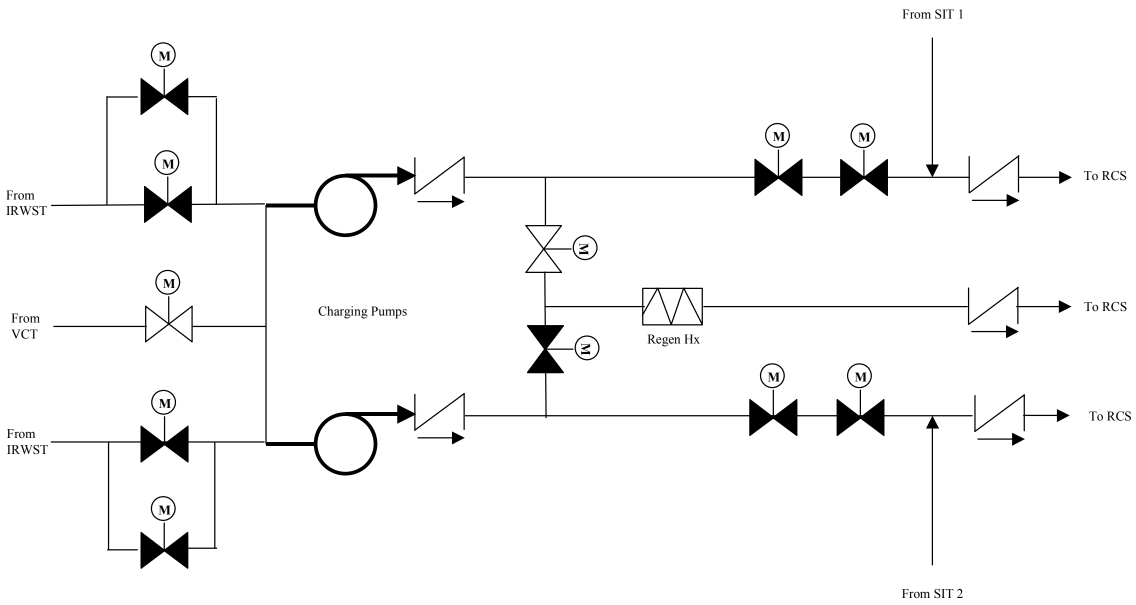

FIG. 3. Advanced conceptual water makeup system schematic.

3. SUMMARY

The de-regulated electric power market requires that the total costs of new nuclear power plants be competitive with alternative forms of power generation. The AP1000 design is based on the design and licensing of the AP600 design, and it is economically competitive in today's market due to its larger power output. This design was developed through use of technical information and PSA analysis; it is based on current U. S. Nuclear Regulatory Commission regulations and is available for deployment now. For the longer-term markets, the U. S. Department of Energy is conducting investigations into the feasibility of new reactor designs, and a new look at the design and regulatory processes is being studied through the Smart Equipment, the DPCIT, and the Risk- Informed projects.

In the Risk-Informed project, the feasibility of a highly risk-informed, or risk-based, design and regulatory process is being investigated. A new process has been outlined which places PSAs and probabilistic methods for addressing uncertainty ahead of the previous deterministic "defense in depth" method of addressing uncertainty. The new process is being "tested" through sample problems and early results indicate that the safety injection function can be combined with the normally-operating charging system; thus, eliminating the need for most of the emergency core cooling system equipment. The new risk-based process will require a new approach to resolution of uncertainty, and it is expected that new regulations will have to be developed.

Successful completion of the Risk-Informed project will not only provide design and regulatory process methodologies that are applicable to today's ALWRs, but will also investigate the generic applicability to other technologies such as the Pebble Bed Modular

Reactor. The end result of this and other NERI programs is expected to be nuclear plant design options with capital costs lower than $1000/KW.

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Expected benefit from new approach for equipment purchasing policy

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Abstract. The demands of the electricity market have led the French nuclear industry to seek ways of reducing the purchasing costs nuclear power plant components while maintaining an equivalent level of quality and safety. A project (ECOREP) was undertaken on this subject from 1997 to 2000 by EDF and FRAMATOME. Based on in-depth exchanges with suppliers and major companies in non-nuclear sectors, it highlighted the over-specific and sometimes unjustified nature of many of the existing requirements existing. Covering the areas of both general instructions (quality assurance, supplier surveillance, documentation) and technical requirements, the project recognised that there was some good practice in non-nuclear industries which could be adopted without any adverse effects, particularly by ensuring that the requirements are defined in terms of ends rather than means. The simulation analyses carried out led to an estimate that a saving of at least 10% is feasible by attempting to emulate this good non-nuclear industrial practice wherever possible. In particular, reference to more widely-used conventional manufacturing codes is recommended.

1. APPROACH ADOPTED

1.1. Reasons

In the context of liberalization of the electricity market and savings on the costs of other systems (gas, coal), it is important for an operator such as EDF and a manufacturer such as FRAMATOME:

— to ensure the competitiveness of the current nuclear production facilities by controlling their maintenance costs,
— to reduce the construction costs of the plants,

in order to keep the nuclear option open until the units currently in service in France are due for replacement and to benefit export bids.

On the particular aspect of equipment purchases, it has been apparent for a number of years that:

— the over-specific requirements of the nuclear industry are less and less justified in the context of industry in general, which has achieved very satisfactory levels of quality and technical performance;
— with this over-specification, the purchasing costs rise higher as the order volume falls;
— some established nuclear suppliers have disappeared due to the difficulty of meeting requirements which are too specialized for markets of limited size.

Following a proposition by FRAMATOME to EDF at the end of 1996, an operation aimed at developing the requirements of the nuclear industry in France to move closer to standard industrial practice in order to reduce equipment purchasing costs was prepared during the first quarter of 1997.

The planned operation was in addition to other action initiated to improve overall plant performance, particularly on the subject of design within the scope of the EPR project.

1.2. Objectives and principles

The ECOREP (ECOnomie des Réacteurs à Eau Pressurisée) project was launched in April 1997. Its objective was defined as follows:

> *"(to conduct) surveys with the intention of proposing changes to the practices or instructions applicable to the manufacture of the mechanical equipment (mainly outside the main primary system and main secondary system) and electrical equipment for the nuclear islands of PWR plants, with the objective of reducing the investment costs while maintaining a level of quality and safety equivalent to the previous production".*

Therefore the ECOREP project was not concerned with the total costs of ownership of the equipment, which also includes the maintenance over the life of the components, but only with the purchasing costs of the equipment.

The following was also noted:

> *"This work must conclude with propositions for changes which can be applicable as a priority to the EPR project and if possible to the engineering and purchasing of spare parts for the plants in operation and to any export contracts".*

The areas decided on for examination and proposed amendments to the requirements are in two main categories:

- Relationships with suppliers on Quality Assurance, production surveillance and documentation required;
- Technical instructions covering the design (dimensioning, technological options) and production of the equipment (purchasing, manufacture and inspection) contained in the applicable system of reference consisting of the RCC-M and RCC-E French nuclear manufacturing codes and the EDF and FRAMATOME technical documentation.

When the project was launched, some strong ideas were put forward regarding the principles that should govern the work to be carried out:

- To ask some fundamental questions so that the validity of the requirements and the effectiveness of the practice could be evaluated with complete objectivity;
- To enter into a dialogue with suppliers in order to reveal the reasons for additional costs, listen to any proposition for improvements and analyse the consequences by an iterative approach;
- To take account of developments in conventional industries on Quality (ISO system of reference), standards and new technologies.

In relation to the methodology, it was decided at the start of the project on the basis of these principles to favour obtaining a good knowledge of industrial practice outside the nuclear sector by surveys on the ground and targeted exercises, in order to facilitate adaptation of the existing practices that go back to the era when France's nuclear facilities were built.

1.3. Project sequence

Nearly 200 people were involved in implementation of the project, representing more than 60,000 hours' work.

The project was divided into 3 main phases:

Phase 1 (April 1997 - April 1998)

The objective of the first phase was to produce the report on current practice compared with practice in related industries.

It was carried out by means of surveys among 32 established suppliers to EDF and FRAMATOME in order to identify with them:

- The systems of reference used for production of equipment in sectors other than the nuclear industry;
- The differences between current requirements in the nuclear industry and other industrial sectors;
- Possible ways of improvement.

The conclusions from this first phase helped to identify the subjects on which to base formation of the EDF/FRAMATOME working groups responsible for examining the changes to be proposed.

Phase 2 (January 1998 - December 1999)

The objective of the second phase of the project was preparation of the modification bids by the EDF/FRAMATOME working groups, consisting of 14 agreed subjects for study, including the following:

- Quality Assurance
- Supplier surveillance
- Documentation required from suppliers
- RCC-M nuclear code for mechanical equipment
- RCC-E nuclear code for electrical equipment
- Design and technology of pumps and valves
- CODAP and CODETI French code for conventional mechanical manufacture
- Design and technology of electrical equipment
- General electrical installation and cabling.

Phase 3 (February - December 1999)

The objective of the third phase of the project was to evaluate the suitability of the proposed changes and quantify the resultant savings on a representative sample of equipment which has the highest share of costs in relation to the total budget for a unit, by means of consultations among selected suppliers, based on documentation containing a summary of the results of the surveys.

1.4. Comparison with conventional industry

The knowledge of industrial practice outside the nuclear industry was obtained by two complementary approaches.

The first was to have discussions with the suppliers visited during the surveys in phase 1. These suppliers are all currently operating outside the nuclear industry and the supply of nuclear equipment generally represents a very small share of their turnover. These companies all joined in the benchmarking game by explaining their normal practice with other customers.

The second approach was to contact economic players in other sectors of industry directly. Exchanges then took place with:

- "Key account" customer organisations such as FRANCE TELECOM and the Government Armament Department; both have had to deal with the problem of "over-specification" and their experience of standardization proved most interesting;
- An engineering company, TECHNIP;
- Industrial companies such as AEROSPATIALE Aircraft and GIAT Industrie.

The work of the US Department of Defense in relation to its procurement standardization operations (MILSPEC REFORM) was also analysed.

2. CHANGES PROPOSED BY THE ECOREP PROJECT

2.1. General

The ECOREP changes were designed as a coherent package because their objective was to simplify both the requirements in terms of relationships with suppliers and the technical specifications. The way in which they were formulated on a given theme always made allowances for possible intrusion into the other theme. Their objective was to make the declared intention of emulating the standards in non-nuclear industries so as to maintain an equivalent level of quality clear to and credible for suppliers. Consequently, what was presented to the companies was an ECOREP package and this was the basis on which the consultation exercises with them in phase 3 of the project took place and on which they prepared their numbers.

2.2. Changes in relation to general instructions

2.2.1. Quality Assurance

The main change involves redefining the requirements for the Quality Assurance system by reference to the basic standard ISO 9000 with which the great majority of suppliers already conform. For products meeting safety requirements, the new system of reference to be followed therefore consists of the following documents:

- Standard ISO 9001 or 9002 (depending on supplies);
- IAEA code 50-C-SG-Q;
- "Quality" regulation dated 10/08/84 (for France).

The wording prepared is simpler and clearer than the previous versions; they incorporate comments from the suppliers and feedback from application of the current versions.

This ECOREP approach obviously means that the EDF and FRAMATOME audit teams take into account the existence of ISO certification by suppliers, which in practice reduces the length of the audits.

2.2.2. Supplier surveillance

The ECOREP changes on this subject are significant; they allow a less burdensome presence by inspectors at suppliers' premises while retaining a level of surveillance which is at least as effective as the present one and generating financial savings.

These changes are as follows:

1. To adapt the surveillance operations to suit the level of criticality of the product covered by the order;
2. To adapt the surveillance operations to suit the level of confidence in the supplier;
3. To carry out surveillance operations mainly on the activities relevant to quality. The emphasis has to be placed on what is important during processing of orders by suppliers;
4. To give the supplier reliable information at the invitation to tender stage on the level of surveillance to be implemented;
5. To relax the procedures for intervention by the inspectors at suppliers' plants in order to impose fewer constraints in the process of production of the supplies ordered by allowing maximum flexibility of manufacture;
6. To relax the documentation requirements associated with the surveillance operations by more flexible documentation transmission methods.

As a continuation of the work of the ECOREP working group and with the same objective of reducing costs, exchanges are ongoing between EDF and FRAMATOME to optimise the work of inspectors at suppliers' plants at a general level.

2.2.3. Documentation required from suppliers

The changes on this subject are also important; they result in a significant relaxation of the requirements from the supplier in both quality and quantity.

Fewer documents are required from the supplier (some documents considered unnecessary are eliminated) a large number of those still required are no longer to be transmitted but just kept available at the plant where the order is produced. As with the surveillance, these documentation requirements are adapted:

— to suit the criticality of the product concerned; the levels of criticality defined for the surveillance are used;
— to suit the level of confidence in the supplier.

The format requirements for the documentation are largely discontinued to enable companies to use their standard documents in the existing formats.

A move to electronic document transmission is specified (it is allowed but not compulsory).

2.3. Changes in technical requirements

The basic principle adopted was to limit the requirements to those relevant to the ends and allow maximum initiative to suppliers in relation to the means, in other words their know-how.

Some examples of simplification of requirements are given below:

2.3.1. RCC-M French code for the manufacture of nuclear mechanical equipment

The changes have led to numerous amendments being adopted on 4 main subjects:

- Practice simplification;
- Improvements in purchasing conditions;
- Practice standardization;
- Incorporation of changes in standards.

2.3.2. Design and technology of pumps and valves

Many simplifications of the technological requirements currently defined in the technical specifications have been adopted, together with application of the API 610 code for certain aspects of pumps in order to make them less specific for nuclear equipment.

2.3.3. RCC-E French code for the manufacture of nuclear electrical equipment

The general principles adopted are as follows:

- RCC-E focussed on the requirements for safety classified equipment;
- RCC-E clearly differentiating the requirements in terms of design (engineering) from those dealing with manufacture (industrial companies);
- General application of standards (CENELEC, CEI, ISO).

2.3.4. Selection of electrical fittings and accessories

The change consists of implementation of a new strategy for purchasing of electrical fittings and accessories to replace exclusive use of the current Lists of Electrical Fittings and Accessories (LPM) in which equipment and suppliers are specified.

2.3.5. Design and technology of electrical equipment

The standard Technical Specifications (CST) were the subject of amendment to meet 3 main objectives:

- Emulation of conventional industrial practice;
- Reference to standards (CEI, ISO, CENELEC) whenever possible;
- Rationalization of the wording.

2.3.6. Diesel-generator sets

The approach was to compare with a supplier the differences between the conditions of manufacture of a diesel-generator set from standard production with a performance which is well suited to the needs and the actual conditions of manufacture of equivalent equipment for the French nuclear fleet.

2.4. Use of conventional manufacturing codes

In the surveys in phase 1, the companies operating in the boilermaking and piping sectors unanimously recommended abandoning the RCC-M nuclear code for equipment outside the Reactor Coolant System and Main Secondary System and moving towards conventional regulations, particularly the two French codes CODAP (pressure vessels) and CODETI (piping).

A comprehensive comparison of the CODAP and CODETI and the RCC-M requirements was carried out. It was supplemented by special draft preparation exercises on the basis of these codes and specifications for the supply of heat exchanger, tank and pipe batches. Inquiries were made to six companies with these specifications to evaluate the feasibility of the change of reference regulations and its impact on the associated equipment costs.

The results of all the analyses carried out are given below:

2.4.1. Fabricated components

- For these components, it is possible for the specification accompanying design and manufacture in accordance with CODAP to include the specific additional requirements which are imperative in order to allow for the special nuclear characteristics. These specific requirements are not in conflict with the other CODAP requirements;
- Manufacture of these components on the basis of CODAP and the nuclear requirements results in a very considerable saving on the equipment production cost of over 20%. A clear break with existing practice certainly appears to be a major factor in achieving the objective of reducing costs;
- The scope allowed to the companies in terms of design on the basis of the functional requirements alone showed that interesting technical/economic solutions can be implemented. An additional saving again exceeding 20% was thus demonstrated;
- Good practice for these components, and probably for many others also, requires inquiries to be issued with a relatively open specification so that technically interesting solutions are not ruled out, and the additional requirements then to be optimised according to the responses obtained and the choice of manufacturer;
- The CODAP exercise shows that the option of moving towards conventional mechanical manufacturing regulations is quite feasible. This principle is one of the main benefits of the ECOREP project; it merits being developed in further action to establish the generic rules of the move and enable it to be extended to other international code.

2.4.2. Pipework

As far as pipework is concerned, the use of CODETI is conceivable for relatively unstressed systems with the addition of some supplementary requirements (control of purchases and fabrications, types of welded joint, etc.).

For systems with greater stress performing safety functions, far more additional requirements are necessary. It is preferable to retain the RCC-M code or apply other widely used standards such as ASME III or ANSI B 31.1.

Nonetheless, the exercise resulted in an estimate of savings achieved of 12.4% on the piping item (purchasing + installation outside level 1) for one unit by the use of CODETI with the necessary additional requirements.

3. SAVINGS OBTAINED BY ECOREP

In phase 3 of the project, the consultations with suppliers and the studies by the experts involved in preparing propositions for changes to the requirements formed the basis used by the evaluation group set up to estimate the savings achieved by ECOREP.

A saving of between 10 and 12% on the purchasing costs of the relevant equipment for the nuclear island in a series plant was identified.

Further analysis of the saving shows that the work under ECOREP helped to identify 2 sources:

- Savings on the production of equipment of identical design;
- Savings due to re-examination of the design.

In terms of lead time for factory production of the equipment, a saving of 3 months on an average lead time of 18 months was also identified. This is achieved by:

- Greater manufacturing flexibility due to relaxation of the surveillance formalities;
- Elimination of some unjustified technical requirements, enabling companies to arrange their purchasing and manufacturing operations in a more standard form.

4. ANALYSIS OF ECOREP RESULTS - COMMENTS

1) The considerations which led in 1996 to the bid to undertake an operation aimed at amending the requirements in our nuclear industry in order to reduce equipment costs have been fully confirmed:

    - Major and often unjustified disparities from current industrial practice were observed;
    - Considerable savings in cost and time can be achieved.

    The proposed changes in requirements enabled significant savings to be identified when the costs for a series unit were estimated:

    - A saving of more than 10% on the purchasing costs of nuclear island equipment;
    - A reduction in the factory production lead times for the equipment which translates into a saving of 3 months on the average lead time of an order of 18 months.

2) The major benefit from the ECOREP project is the awareness of:

- The possibility of relaxing requirements while maintaining a level of quality equivalent to previous production;
- The great advantage of holding very open discussions with suppliers and increasing knowledge of practice in other industrial sectors.

3) The implementation of ECOREP must be considered as a global approach. It was an ECOREP package that was developed and presented to suppliers. The amendments to the requirements must therefore be adopted in their entirety as far as reasonably possible.

4) At the end of the project, sources of additional savings have been clearly identified. The best approach is to go further with the move to use of other mechanical manufacturing codes more widely used than the French RCC-M standards, particularly the ASME American standards (nuclear and conventional parts). The results of the consultations show that the greatest savings are obtained when the changes represent clear breaks with current practice (e.g. use of CODAP for pressure vessels).

5) It is also important to remember that ECOREP is a project based on nuclear island equipment and it would be appropriate to extend the same systematic method to examination of cost reductions on all the other equipment in the plant.

6) Finally, it is worth reporting the conclusions of the recent report "Reducing the capital costs of nuclear plants" (ISBN 92-64-27144-9 – 1st quarter 2000) prepared by a group of experts for the OECD on the subject of purchasing policies:

"In relation to product suppliers and service and work providers, the elements described below are sources of savings.

The quality assurance and quality control programmes represent a major cost item in the construction of a nuclear plant. They frequently result in the creation of extremely expensive "nuclear" components which are actually very similar to components used in non-nuclear environments. These programmes were fully justified in the 1970's when existing industrial standards were less focussed on quality assurance and control and the nuclear industry needed better-quality products. The quality assurance and control specifications applied to standard industrial products today have closed the gap so well that many of these products would meet the quality criteria for the nuclear industry. Large-scale use of non-nuclear components, even in safety systems, would achieve a significant reduction in costs without compromising the overall safety.

It is possible to improve tender procedures whatever the sources of the various contracts (package supplier, prime contractors, client) by:

- *Specifying packages that enable suppliers to offer their standard products;*
- *Preparing simplified tender documents and specifications focussing on the basics so that bidders can reduce their provisions for contingencies and risks ;*
- *Increasing the number of bidders to prevent monopoly niches being perpetuated or established ;*

— *Adopting stricter conditions in contracts (firm prices, all inclusive, satisfactory performance guarantees and penalty clauses, strict rules for design modifications).”*

The report proposes approaches fully in line with those in the ECOREP project: study of standard products – simple and precise order documentation covering the basics, i.e. the objectives – development of competition.

5. CONCLUSIONS

The ECOREP project was initiated in April 1997 and completed in three years. It identified numerous measures for relaxation of the requirements of the nuclear system of reference, resulting in savings estimated at more than 10% being identified for the nuclear island of a series unit.

The savings achieved can be considerably increased by emulating practice in conventional industrial sectors even more closely, particularly by wider reference to their codes and standards.

ECOREP represents a further stage in the battle for competitiveness which must certainly be continued.

The application of an integrated approach to design, procurement and contruction in reducing overall nuclear power plant costs

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Abstract. As part of its on-going efforts to reduce the cost of CANDU nuclear power plants, AECL has embarked on an integrated approach to design, procurement and construction activities associated with new CANDU 6 and CANDU 9 projects. The approach is predicated on the fact there is a vast quantity of information that needs to be managed and controlled over the life of a nuclear power plant project. Therefore, ensuring the completeness and correctness of all the information needed by all project participants, facilitating sharing of this information amongst the project's participants, and automating the various deliverable production processes offers significant potential not only for overall project cost (and schedule) savings but also for reducing operations and maintenance costs once the plant enters service. Facilitating and indeed of key importance to this approach is the use of a suite of integrated information technology-based engineering, procurement and project control tools used throughout the design, engineering, procurement and construction phases of the project. A unique and important feature of these tools is their high degree of integration both from a work process and a data perspective. Use of these tools is well underway on AECL's Qinshan Project which is realizing significant benefits in cost and schedule. This paper will describe the approach AECL is taking, along with the tools it has both put in place, and those additional items planned for the future along with the cost, schedule and quality benefits that arise from their use. Progress to date on the Qinshan project also will be discussed as well as the expected application to the plant once it has gone into service will also be discussed.

1. INTRODUCTION

The delivery of a nuclear power plant such as a CANDU 6 or a CANDU 9 is a major and complicated undertaking. It involves several major participants in the form of project management and engineering firms, procurement organizations, constructors, hundreds of contractors and suppliers and thousands of employees drawn from hundreds of disciplines and trades. The volume of information required to be created, managed and exchanged amongst the participants is also huge, complicated by the fact that the participants are likely situated in geographically disparate locations around the world.

In addition to evolving its CANDU products to meet the emerging design and performance requirements expected by today's operating utilities, AECL also is aggressively attempting to integrate the work processes and information management needs across all project participants as well as across the entire plant lifecycle.

Key to these improved processes is the use of a suite of advanced information technology based engineering and project control tools. The intent in both cases is to reduce the cost and schedule of delivering a CANDU project. As well it is envisioned that using these tools throughout the design, engineering, procurement and construction phases will ideally position the utility to manage and maintain the plant post in-service.

This paper discusses the approach AECL has taken in developing these tools, their use (particularly on AECL's latest project) along with the benefits resulting from their use will be discussed.

2. INTEGRATING PROCESSES AND TOOLS

In order to significantly enhance the engineering, procurement, construction and commissioning activities, AECL has committed to undertak its scope of work in a fully electronic and integrated manner. This is being accomplished thorough the use of an integrated suite of electronic information technology tools customized in-house to comply with nuclear industry and CANDU specific engineering practices and design standards. As a result, the design deliverables produced by these tools follow accepted nuclear and proven CANDU engineering practices and standards creating a design less prone to design errors.

These tools produce information and reports that can be used during the engineering design phase as well as during construction and commissioning and post in-service.

2.1. Development approach

It is generally recognized that using electronic tools, or information technology in general, to replace existing, manual and paper based processes will not immediately yield any significant benefits such as an increase in productivity or the elimination of errors. Indeed in some cases the introduction of this technology can result in an overall decrease in productivity. Benefits must e engineered by examining the details of current work practices to find those steps at which gains can be made.

To ensure the maximum possible benefits, the following approach has been taken for implementation and subsequent utilization of electronic tools at AECL:

- Labour intensive and error-prone engineering, procurement, and other project delivery activities are identified. Tools are then put in place which substantially automate these activities, eliminate the introduction of errors and capture all necessary project information;
- The underlying databases used by these tools are integrated to ensure data are uniquely stored and referenced;
- These integrated databases are made available to all projects participants that require them to undertake their scope of work;
- The ability to produce deliverables such as drawings, lists, bills of materials etc. directly from these integrated databases is implemented wherever possible and feasible;
- Wherever possible use is made of third party, commercially available, supported and proven software;
- The information technology providers, responsible for the implementation and on-going support of the tools, work closely and intimately with project staff who use these tools on a daily basis.

2.2. Tools and databases

A short description of these key tools and their associated databases is given in this Section.

2.2.1. 3D CADDS for plant design

The use of 3D CADD models and systems is well established in the design, engineering and construction of process and power plants. To date, several hundred billion dollars worth of conventional process and power plants, designed using 3D CADDS systems, have been built and are now in operation. The benefits of using 3D CADDS are generally recognized and can be summarized as:

- improved productivity through concurrent engineering, procurement and construction;
- near elimination of rework and interferences;
- automatic drawing extraction and;
- accurate material take-off.

Starting with the Qinshan project AECL is committed to the use of 3D Plant Design model on all CANDU projects. Specifically for Qinshan, Intergraph's Plant Design System (PDS) is being used to build a high fidelity 3D model of the nuclear steam plant. This includes all:

- Engineered piping and in-line piping components (valves, strainers, etc.);
- Raceways;
- Structural steel;
- Concrete;
- Heating, Air Conditioning and Ventilation components;
- Equipment (tanks, pumps, heat exchangers, etc.);
- Piping supports;
- Embedded parts and plates.

Once all interferences are resolved. the following deliverables are extracted from the 3D CADDS model:

- Piping system isometric drawings,
- General arrangement drawings,
- Material take off lists.

In addition to its use in ensuring an interference-free design and the production of the engineering deliverables described above, the 3D model is being used extensively on the project for plant visualization work, walk through and installation planning.

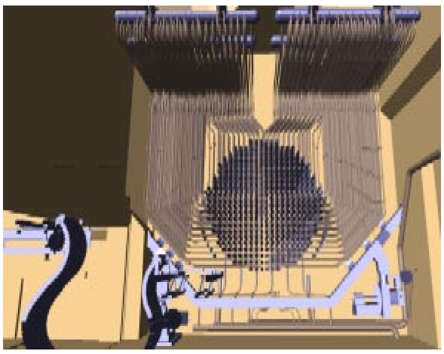

FIG. 1a. Image of reactor face extracted from the Qinshan 3D CADDS model.

2.2.2. Instrumentation and control design

IntEC is an AECL developed tool for wiring and cabling design and information management. It allows engineers and designers to perform design functions such as;

- connecting wires to devices,
- assigning wires to cables,
- routing cables,
- assigning cables to raceways,
- detailed raceway design,
- etc.

in a manner which ensures all CANDU design conventions are fully observed along with the design requirements associated with channelization, separation and redundancy.

In addition to its use in the engineering office IntEC will be made available at the construction site where site personnel will view the information, update it as necessary and extract the various reports such as end-to-end wiring reports and cable pull sheets directly. In addition, through a utility called IntEC-Vision, users can depict critical wiring and cabling information graphically from information extracted from the IntEC database. Currently IntEC-Vision is able to generate;

- End-to-End Wiring drawings,
- Connection by Device drawings,
- Instrument Wiring Loop diagrams and,
- Cable Block Diagrams.

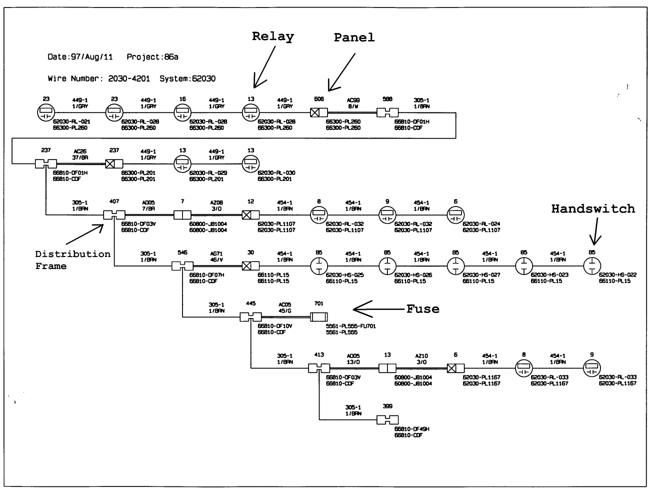

FIG. 1b. Sample end-to-end wiring diagram produced by Intec Vision.

IntEC also manages detailed information on instruments such as their functional requirements, procurement specifications, association to documents such as a fabrication drawing or maintenance procedure, grouping of instruments within loops, etc.

The instrumentation and control and reactor and fuel handing controls disciplines on the project are currently using IntEC-Equipment to produce:

- instrument lists,
- instrument application sheets,
- digital control computer input/output lists and,
- environmental qualification component lists.

2.2.3. 3D CADD for fueling machine modelling

Unlike previous CANDU projects, on the Qinshan Project AECL has taken on full responsibility for manufacturing the fuelling machines. To assist in undertaking this scope, a 3D solid mechanical model of the fuelling machine is being built using. Unigraphic's Solid Edge product.

Solid Edge is a mechanical modeller which allows for modelling at a detailed component level. It allows for mechanical assembly and part modelling, interference allowance and tolerance checking.

Although detailed fabrication drawings are being extracted from the fuelling machine model, some use is also being made of the software's ability to transfer information directly to numerically controlled milling and tube bending machines.

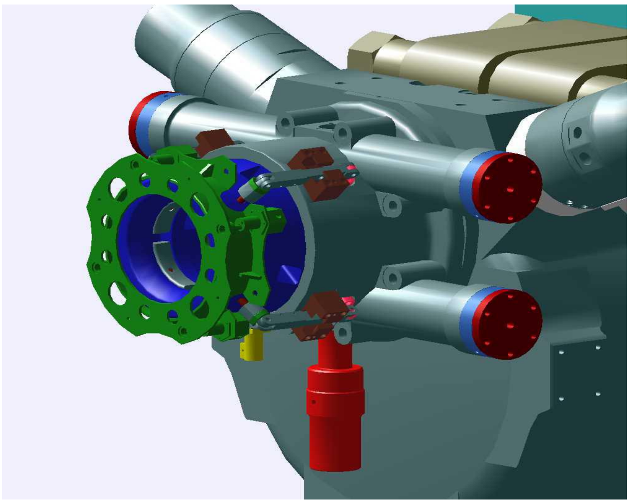

FIG. 2. CANDU 6 fuelling machine component modelled using solid edge.

2.2.4. Materials management

To ensure that required materials are available at the construction site in a timely manner and without unnecessary surpluses and to enhance the process of bundling and providing material to the site construction forces, AECL has adopted a tightly integrated approach to materials management, tracking and control which covers the entire lifecycle of the project; from engineering through the turn-over to operations.

This is accomplished through the use of the CANDU Material Management System (CMMS). CMMS is an AECL developed tool specifically designed to meet the needs of CANDU nuclear projects, including their requirements for nuclear grade materials, code compliance and quality assurance. Previously used on the Wolsong and Cernavoda CANDU projects, the tool has been significantly enhanced for use on the Qinshan CANDU Project particularly in its direct application to the project's procurement and construction needs.

CMMS is based on the project's stock code catalogue, which currently uniquely defines about 60,000 items. Material demand is largely captured by automatic electronic download from other tools used during engineering; most notably the 3-D CADDS plant model. Where and when necessary material demand is also input manually.

Once the material demand is established, CMMS aggregates and groups material to facilitate purchasing. CMMS contains a complete integrated purchasing system, which includes the tendering, purchasing, releasing and shipping functions. These are in turn integrated with receiving of material at the construction site and the maintenance of inventory and issuing of material from the site warehouse. Material is then grouped by area or trade into so called Construction Work Packages, which are then used to define and control the flow of material to the construction forces.

A particularly useful feature of CMMS is that it allows different groups to view information of interest to them in the manner they find most useful. For example, designers can view material information organized by system, constructors can view by plant area or trade, and commissioning staff can view by equipment.

2.2.5. Project scheduling

Primavera's P3 planning and scheduling software is being used by AECL. Level 1, 2 and 3 project schedules are prepared using this software tool. For AECL's scope of work this tool is interfaced with AECL's financial system to facilitate the comparisons with actual expenditures and calculation of earned values.

2.2.6. Deliverables management and control

To manage its deliverables, and other related documents and drawings, AECL has adopted Intergraph's state-of-the-art Asset and Information Management (AIM®) system for document and drawing management.

All project documentation is being electronically stored and managed within the AIM system. This includes:

- AECL generated drawings and documents,
- Vendor and contractor generated documents,
- Formal project correspondence and records,
- Work packages (collections of drawings and documents required to perform some function such as construction or commissioning),
- Releases for construction packages,
- Site quality assurance records.

In many cases the inputting of information and documents and drawings into AIM has been fully automated. For example isometric drawings extracted from PDS, bills of material from CMMS and various deliverables from IntEC such as Instrument Application Sheets are automatically input into the AIM system. Given the large number of these, the increase in productivity is significant.

In cases where the project receives only a paper copy of a document or drawing scanning is used to create an electronic version of it prior to its entry into the AIM system.

In addition to the basic functions expected of any document management system such as managing the information that describes a particular drawing or document i.e. the title, number, author, etc.;

- where the electronic files making up the document are stored;
- revision and version control and;
- user access privileges to the files.

Incorporated into the implementation of AIM are many useful and productivity enhancing features such as,

- the ability to view documents or drawings directly on a users workstation screen regardless of format,
- the ability to plot drawings to any scale,
- the ability to easily plot many drawings at once.

In addition, to totally eliminate the need for formally dealing with paper documents on the project, the ability to electronically sign and place engineers' professional stamps on electronic drawings and documents has been implemented.

Furthermore the functionality of AIM has been extended to support the project's document control process. An electronic tool called TRAK, has been built on top of the AIM product to:

- define the project's deliverables baseline (i.e. documents and drawings to be released to construction, the client or other organizations);
- schedule and manage the release and receipt of deliverables;
- produce various project control reports;
- generate work packages;
- prepare releases, transmittals;
- exchange deliverables and associated information between AIM systems.

3. INFRASTRUCTURE AND INTERFACES

A simplified representation of the information technology infrastructure, i.e. the computers, operating systems, networking technology, database management systems, etc., existing within AECL's main engineering and procurement offices is shown in figure 4. In order to ensure nuclear steam plant design and procurement work on the Qinshan Project is carried out in a highly integrated fashion, AECL's subcontractor has been given access and is using, as appropriate to their scope of work, the same tools and databases as AECL, specifically PDS, AIM/TRAK, CMMS and IntEC. With the subcontractor's offices located almost 300 miles away from AECL's main offices this has required the implementation of a high-speed telecommunications link between the two. As well, in order to simplify the interfacing issues between the nuclear steam plant and the balance of plant designers in the instrumentation and control area, the subcontractor has been given a high-speed remote telecommunications link and access to the IntEC suite of tools.

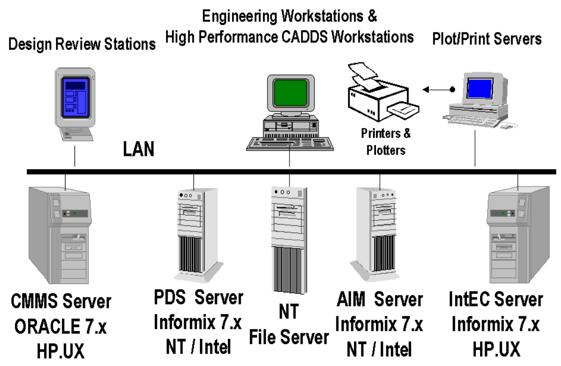

FIG. 3. Information technology infrastructure.

As many of the electronic tools are also used at the Qinshan construction site, a very similar infrastructure to that depicted in figure 2 is also installed there. This allows information to be transmitted back and forth between AECL's office in Canada and the Qinshan construction site in electronic form. Internet based communications technology has recently been put in place for use when a relatively small amount of data is to be transferred.

In addition to the benefits associated with electronically transmitting information to site, having these tools on site and available on the LAN allows the plant owner, AECL and the various site contractors to share and use the same information to manage and carry out their work.

Although the above discussion refers directly to the Qinshan Project a similar approach to the information technology infrastructure will be adopted on all future projects.

4. FUTURE PROJECTS

Future CANDU projects will build on the experiences gained, the information gathered and the tools developed and implemented on the Qinshan Project and during the design phase of the CANDU 9 product.

In addition, as part of its product development program and in anticipation of future CANDU projects, AECL is aggressively extending the degree of automation amongst many of the remaining labour intensive work processes and furthering the level of integration amongst the underlying databases associated with both existing and emerging tools.

For example, it is planned to extend the use of IntEC-Equipment to handle all equipment in the plant and to develop it into a full equipment specification tool. AECL also plans to extend the use of AIM and TRAK to manage and control other project information such as analysis data.

A example of totally new work currently underway, that will be of significant importance to the next CANDU project is the development of an integrated design, analysis and modelling tool for engineered piping supports.

Due to the large number of such supports in a CANDU plant, the time required to design and analyze them and the need to prevent space allocation conflicts, a totally integrated design, analysis and modelling tool is very desirable. AECL is developing such a tool, called the Piping Support Design System (PSDS), as a rapid design, analysis modelling system and data management and documentation system for the supports design. It interfaces with the 3D CADDS model of the plant. It has the capability for design of both the secondary steel as well as catalogue components, for nuclear-class and non-nuclear class code specifications. In addition to the above tasks, PSDS imports pipe stress analysis results to create a database of support design parameters. It will also directly generates fabrication drawing and bills of material. Interference detection with the rest of the plant will be facilitated through a support envelope file interfaced to the 3D CADDS model.

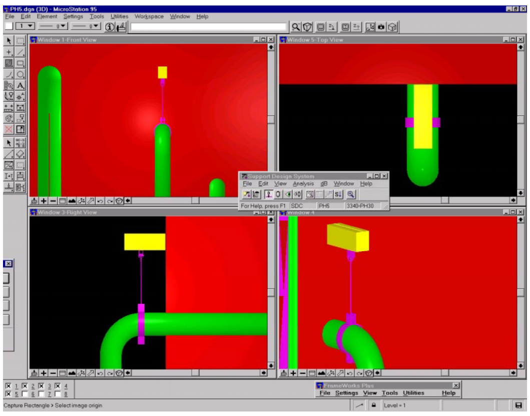

FIG. 4. Sample piping support designed by the piping support design systems.

5. CONCLUSIONS

An integrated set of engineering tools supporting a set of integrated work processes have been developed and is in use on AECL's Qinshan CANDU project and is planned to be used on all future CANDU projects. Further developments of this suite both in terms of integration and functionality are underway.

These tools, by ensuring the correctness and completeness of the information needed by the project, facilitating sharing of this information amongst the project's participants and by automating various deliverable production processes are having a significant beneficial impact in reducing overall project costs and risks both directly and through reductions in the construction schedule.

In addition to its value during the engineering, procurement, construction and commissioning of the plant, the information gathered by the tools described in this paper will be a valuable aid to operating, maintaining and managing the configuration of the plant once it enters commercial service.

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New technologies for lower-cost design and construction of new nuclear power plants

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Abstract. Electric Power Research Institute studies indicate that in order to be competitive with gas-fired electric power plant capital costs, , new nuclear plant capital cost in the USA must be decreased by at least 35% to 40% relative to costs of some Advanced Light Water Reactors designed in the early 1990s. To address this need, the U. S. Department of Energy is sponsoring three separate projects under its Nuclear Energy Research Initiative. These projects are the Risk-Informed Assessment of Regulatory and Design Requirements for Future Nuclear Power Plants, the Smart Equipment Nuclear Power Plant Program, and the Design, Procure, Construct, Install and Test Program. The goal of the Design-Construction program is reduction of the complete nuclear plant design-procure-construct-install-test cycle schedule and cost. A 3D plant model was combined with a construction schedule to produce a 4D visualization of plant construction, which was then used to analyze plant construction methods. Insights include the need for concurrent engineering, a plant-wide central database, and use of the World-Wide WEB. The goal of Smart Equipment program is to design, develop, and evaluate the methods for implementing smart equipment and predictive maintenance technology. "Smart" equipment means components and systems that are instrumented and monitored to detect incipient failures in order to improve their reliability. The resulting smart equipment methods will be combined with a more risk-informed regulatory approach to allow plant designers to (1) simplify designs without compromising overall reliability and safety and (2) maintain more reliable plants at lower cost. Initial results show that rotating equipment such as charging pumps would benefit most from smart instrumentation and that the technique of Bayesian Belief Networks would be most appropriate for providing input to a health monitoring system.

1. DESIGN, PROCURE, CONSTRUCT, INSTALL AND TEST PROGRAM

Westinghouse and Duke Engineering and Services (DE&S) are participating in two programs aimed at developing advanced technologies to reduce design, procurement, construction installation and testing (DPCIT) costs for future nuclear plants. The first program, under sponsorship from the Electric Power Research Institute (EPRI), is leading a team of industry designers to develop a 4D visualization of its construction plans for the System 80+^{tm} advanced nuclear plant (ANP) design. Construction plan key elements were benchmarked with a side by side visualization of an actual Korean Standard Nuclear Plant (KSNP) construction plan currently being built in the Republic of Korea. The 4D application provided visual assurance that improvements in productivity, product and process proposed for the advanced nuclear plant and/or being used in Korea were readily achievable for implementation.

The second program is being conducted by a team of industry members as part of the U. S. Department of Energy's Nuclear Energy Research Initiative (NERI)[1] [2]. This work is focused on examining the DPCIT cycle as a means of applying new technologies to reduce the cost in all phases of the DPCIT cycle. This effort has identified several methods for improving the DPCIT cycle time and reducing corresponding costs.

1.1. 4D visualization model development

As the 4D visualization model was developed, the tools and methods used were evaluated for their usefulness to nuclear plant projects. An initial functional specification for a 4D application was prepared and various software alternatives examined. A software application, currently under development, from Construction Systems Associates, Inc. (CSA) was selected to provide linkage between the System 80+ 3D-computer model (used as an example) and the construction schedule database. With this linkage, a time dependent computer screen display of the System 80+ 3D model was generated. That display, or 4D visualization, was then used for evaluation of construction sequences to flush out construction plan problems, correct the problems, refine the construction plan, and explain the overall construction plan. "Snapshots" of the 4D construction sequence are shown in Figures 1-4.

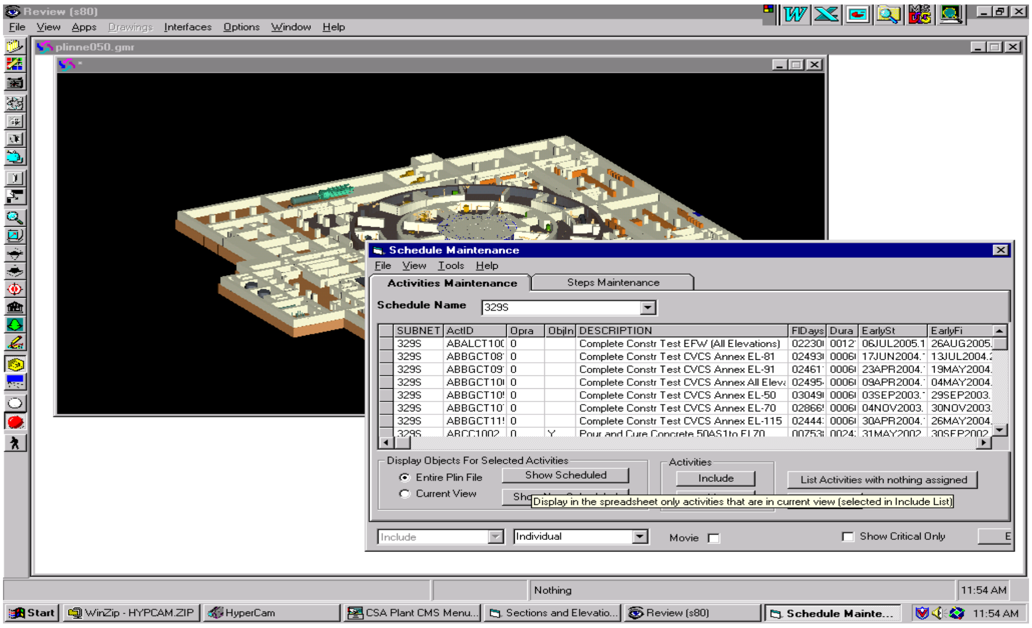

FIG. 1. Snapshot A of system 80+ 4D construction sequence.

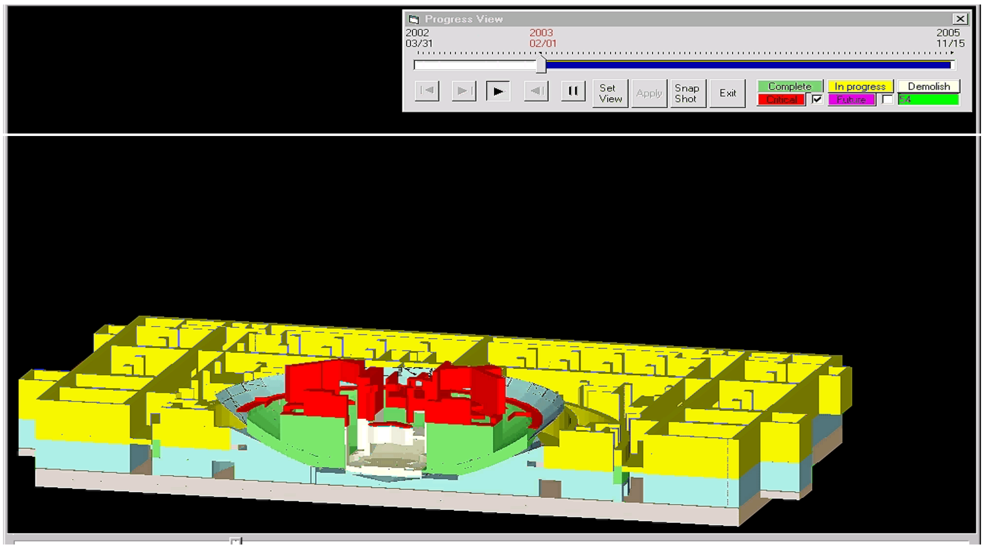

FIG. 2. Snapshot B of system 80+ 4D construction sequence.

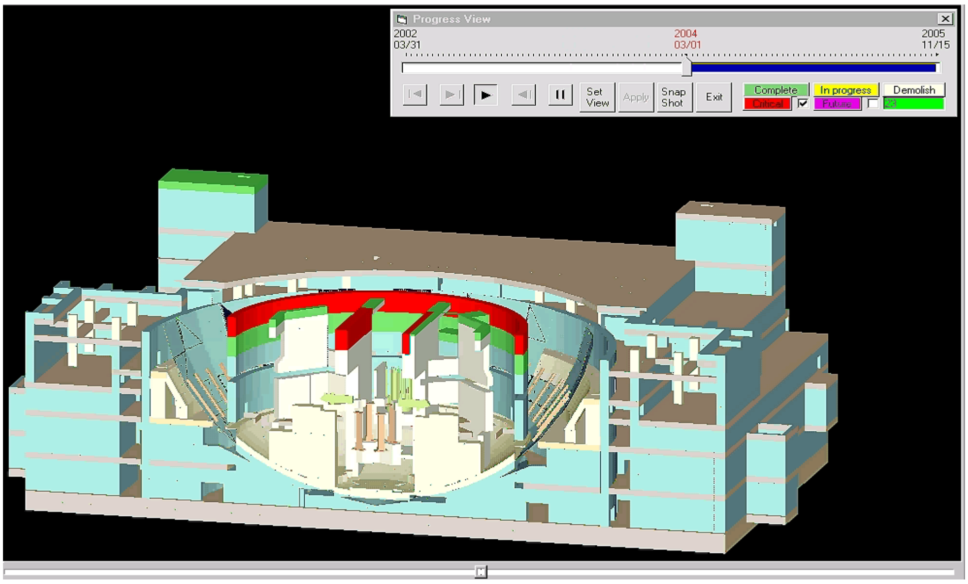

FIG. 3. Snapshot C of system 80+ 4D construction sequence.

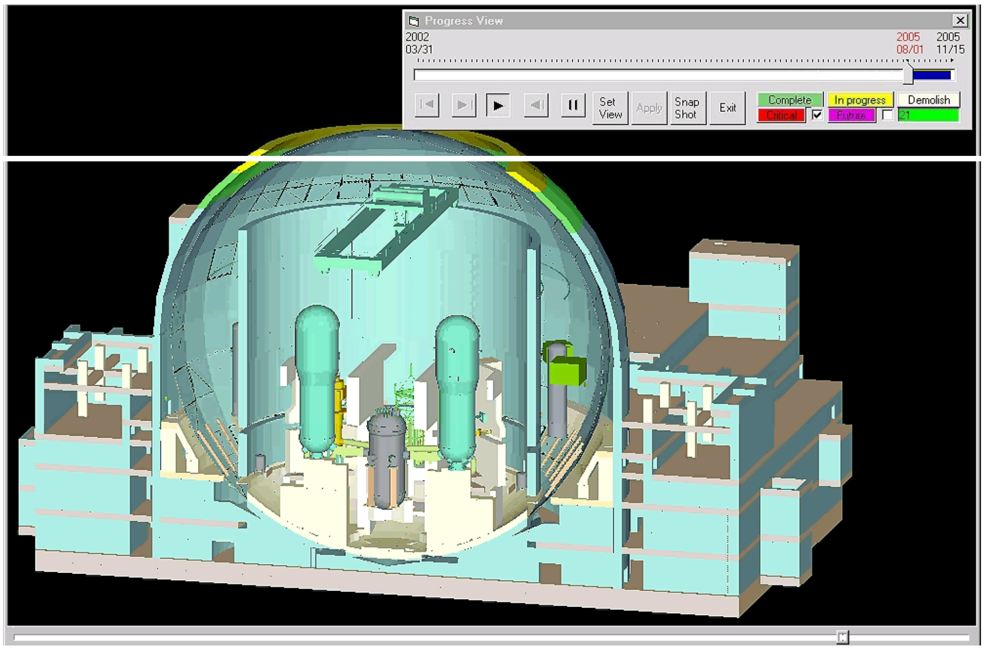

FIG. 4. Snapshot D of system 80+ 4D construction sequence.

Westinghouse was able to identify and correct logic errors in the System 80+ schedule, isolate and evaluate critical construction and installation sequences, and verify that its overall System 80+ schedule was reasonable. The result of this effort was a ~2 month improvement in the construction schedule.

The construction schedule was developed from the high level System 80+ schedule, which was itself developed for a commercial proposal. The major milestone schedule for the System 80+ project was compared with the actual milestone completion durations for the Ulchin 3&4 KSNP being constructed in Korea. The construction durations for these projects compared favorably because the design for the large components and the need to pour the concrete to support these components is on the critical path for both designs. Although the containment and support building designs are different, interior concrete pours for major NSSS component supports define the critical path. So, using actual construction experience from the Ulchin 3&4 construction project and the 4D construction sequence visualization application, the reference System 80+ nuclear island construction and startup schedule has been benchmarked.

1.1.1. Schedule development method

The original System 80+ construction schedule was developed at Level 2 detail in Gantt Chart format. To supplement this schedule, a detailed Level 3 schedule was developed for the Nuclear Island as part of the four-dimensional visualization effort summarized above. The schedule was developed using Primavera's Suretrak Project Manager® software. Including Level 1 milestone and Level 2 summary activities, the schedule includes approximately 3,600 activities for construction and startup. This construction schedule was developed on the basis that as the concrete and structural work in an area was completed, the equipment installation would be started in the completed area. The concrete and structural work would continue at the higher elevations of the plant while equipment was being installed at lower levels. A number of important decisions affecting the construction sequence were incorporated in the schedule.

– Large equipment is installed Over-the-Top and placed in approximate position before the room is closed from above by on-going structural work;
– Pre-fabrication of the Steel Containment Vessel (SCV) in rings adjacent to the containment building and setting of the rings using a large capacity mobile crane;
– Pre-fabricated liners for pools and lined concrete tanks used as the concrete forms;
– Multiplexed instrumentation signals to decrease the amount of cable and terminations.

Construction tests of installed equipment and systems are performed as the equipment is energized by level from the bottom of the plant up, depending on the isolation points within the fluid systems. Startup testing is performed on a system basis as the equipment is turned over from construction to the startup organization. The non-Nuclear Island owner supplied equipment that is required for startup, such as the switchyard and transformers, auxiliary steam, and de-mineralized water production equipment, was included in the schedule to define the required need dates.

The original overall schedule, from the beginning of the project (contract award) to commercial operation, was 77 months and the period from first concrete to fuel load was 49 months. The resulting schedule after review and modification for the 4D demonstration is 74.5 months from beginning of project to commercial operation and the period from first concrete to fuel load is 47 months.

Use of the 4D model increases confidence in the construction schedule because the 4D application's ability to visually identify major out-of-sequence activities and display omitted model objects was used to correct schedule errors. The following are examples of how the 4D visualization application's capabilities were used:

– When the construction progress is displayed in intervals over the construction period, the schedule reviewer can identify incorrectly scheduled items. One example concerns the secondary concrete containment. The containment was divided into construction schedule objects using schema. When the construction sequence was displayed, one ring of concrete was left out such that the upper half of the building appeared to be floating in air. The error was easily identified and corrected;

- Other applications of the 4D visualization function to determine out-of-sequence activities included the civil construction progress of the annex building relative to the containment building. The display clearly indicated that the annex construction was proceeding too rapidly compared to the containment building. This mismatch in progress was not identified in reviews of the schedule. The display alerted the scheduler to a problem and helped direct the review to the incorrect logic;
- The 4D visualization identified a number of instances where the equipment was placed before the supporting floor was in place;
- The 4D visualization of the entire Nuclear Island is complex and difficult to review for completeness and correctness. Detailed reviews are best performed on a portion of the Nuclear Island. The 4D visualization software allows the scheduler to create a sub-schedule of selected activities. The sub-schedule is a powerful tool for reviewing a specific sequence or system for correctness. As an example, a selection of the Reactor Coolant System main loop components identified that due to incorrect logic installation of one of the hot legs was out of sequence. Additionally, one of the cold-leg work packages was misidentified so that the model object was not associated with the corresponding schedule activity. Using the traditional scheduling methods, these errors could only be discovered with difficulty.

1.1. DPCIT cycle

Information Technology (IT) application has transformed a number of industries by making new types of collaboration possible as well as streamlining a number of design and information retrieval processes. The development of Internet technology has enabled substantial reductions in design-procure cycle time for a range of industries. In the recent years, there has been a merger of design and configuration management software such that new designs can be built with much less time and fewer errors. Examples of technological developments in other industries that are being evaluated include the following:

- Design for constructability
- Electronic procurement
- 3D modeling
- Product Data Management Systems
- Enterprise data management software
- Modular design approaches
- Systems dynamics in managing complex projects
- Data driven process modeling.

The DPCIT project team is identifying and evaluating innovative technologies to reduce design, fabrication and construction costs for future nuclear plants. The project's focus is on examining the DPCIT cycle as a means of developing strategies for applying new technologies, since the capital cost of a new nuclear unit is substantially affected by the work practices of the entire cycle. Reduction of DPCIT costs and concurrent reductions in plant construction time are essential in lowering the capital costs of future plants to ensure a competitive environment for nuclear power plants as a viable electrical generation technology. There are a number of innovations from within the nuclear power domain, as well as a number of business practices found in large scale manufacturing operations, to merge into a new DPCIT cycle to reduce capital cost. These strategies will be grouped into the following areas:

– Passive Design And Risked Based Simplification
– Increased Efficiency And Power Up-rate
– Improved Manufacturing Technologies
– Improved Supply Management And Construction Technologies
– Application of Information Technologies.

1.1.1. Cycle improvements

While IT is identified as a strategic area by itself, a common opportunity for IT exists throughout the DPCIT cycle – that is, use of specialized software models for proposed change testing, visualization and validation. The models described below capture the strategies to evaluate the collective benefit and risks of shifting to new technologies.

1.1.1.1.Product model

This model describes the structures and equipment in the plant and their physical relationships. Many of the delays and causes of rework in both design and construction result from interfaces errors between the structures and equipment. Electronic aided design is becoming common place for component and equipment design, as well as plant layout and structural design. The ability to integrate equipment and component models into the plant model is an area that requires improvement to achieve the maximum benefit of electronic aided design.

Design of a nuclear power plant involves many individuals, in diverse organizations, located throughout the world. It can be assumed that electronic models will be developed using different software packages. This is a benefit because participants continue to work with the tools that they have experience with and are best suited for their particular design activity. Using the existing tools also reduces unnecessary investment in common software for the project. However, the ability to share up to date design information among the project participants is key to reducing the design phase of the DPCIT cycle while ultimately eliminating design and construction rework.

Using the component and equipment models developed by the equipment designers in the Product Model is preferred to redoing the models in the plant model software. Interface data is more completely defined by the equipment designers. Time and potential errors are reduced by eliminating the need to redraw the models.

The document database associated with the plant model serves as the directory for and link to the design and construction documentation. The database also serves as the means for sharing other engineering models among the project participants. Using the document database in the plant model, links are developed to project management, supply management and construction management data required for the execution of the project. By associating inspection and installation data recorded during construction with the product model database, the data are stored in a manner that it is readily available during plant operation and maintenance.

With all of the plant information readily available through the Product Model, a construction supervisor can assemble all the information (drawings, procedures), verify material availability, plan and schedule material movement, identify and locate special equipment, and record work package completion status

The construction view of the plant is different from the engineering design view. The construction personnel view the plant by areas and rooms: the design personnel look at the plant in terms of systems and complete structures. The three-dimensional models built for design functions are not suitable for construction schedule viewing. A critical aspect of construction visualization is depicting the three-dimensional engineering design model so that the construction sequence is understood. It is highly desirable to transform the engineering design model into a construction model without requiring the design model elements to be modeled based on the construction sequence. This allows designers to work with a format most convenient for them while allowing the construction staff to work with the same model in a form that best satisfies their unique needs. The ability to keep the construction and design models synchronized as the plant engineering evolves is of paramount importance.

Integration of plant schematic diagrams, such as P&ID's and wiring diagrams, will assist construction and start up activities. Progress on construction can be recorded using the visualization tool linkage to the schedule. The status can then be displayed on the schematics to monitor system completion. Start up planning can be performed using the schematics to identify system and sub-system testing. The linkage to the plant model allows display of the equipment location in the plant. This ensures the physical location of an item is known prior to commencement of a test. Any special access requirements are identified and planned.
Some of the insights from the program's first year include:

- Reduce rework by having designers and builders working together in Design/Build Teams;
- Simplification of ALWR designs by cutting out 50% of the Safety Related Equipment will not achieve commensurate cost reduction and schedule reduction goals;
- Evolutionary ALWR emphasized O&M controls versus capital cost control which resulted in larger footprint of safety related structures and thus higher cost per square foot of constructed volume;
- A large percentage of space in System 80+ safety class buildings is not used for Safety Related equipment;
- Complete design is needed early (all Safety Related equipment and anything >1" needs to be run) with complete BOM;
- Current work on bulk materials indicates that the large mass of concrete/rebar placement is very limiting.

1.2.1.2. Productivity model

In order to evaluate the DPCIT cycle, it is necessary to understand the activities conducted during the cycle, and their relationships to each other. The activities and their relationships to each other depicted as a logic network schedule is the Productivity Model. The logic network permits assessing the project's critical activities to identify schedule reduction alternatives. These alternatives can include changes to the activity relationships and durations, changes to the design to mitigate barriers to the completion of construction (Product Model), and improvements in the methods used to perform the required activities (Process Model). As the changes are implemented in the logic network, the effect of the changes can be evaluated relative to the initial (baseline) schedule. The evaluation provides the means to measure the impact of the improvements.

1.2.1.3. Productivity model and product model integration

The Productivity Model uses scheduling programs such as Primavera to represent the DPCIT cycle activities, their duration and the associated resources. Improvements in methods that simplify activities or reduce resource demand are illustrated and evaluated in this model. In the EPRI program discussed in the first section of this paper, the Productivity Model is linked to the 3D Product Model to create 4D visualization. Some insights from this effort include:

– Need an overall site management plan for movement of material, modules, people and temporary services as well as equipment lifts;
– Work package level of detail and accomplishment remains unexamined as far as substantial improvements;
– Schedule acceleration increase weather vulnerabilities as well as possible labor disruption impacts.

Four-dimensional visualization unifies the Product Model (three-dimensional plant design model) with the Productivity Model (construction schedule) to permit construction sequence visualization. Construction department periodic reviews during plant design and construction schedule development provide the input necessary to ensure the plant can be built in a cost-effective manner. The practicality and completeness of the schedule can be verified well in advance of construction commencement. Alternate construction sequences can be generated in the four-dimensional visualization to evaluate cost and schedule savings. The visualization allows the trial run of the alternate construction sequence prior to its execution in the plant.

1.2.1.4. Process model

The Process Model defines the overall DPCIT cycle as a combination of processes, each of which represents opportunities for improvements. Additionally, efficient execution of the DPCIT cycle and its processes is the challenge of complex process project management. Complex process project management is characterized by extensive process interrelationships and positive feedback loops. Recently, failure to manage these interrelationships and feedback loops has resulted in large project overruns and missed schedules for a number of complex civil and power plant construction projects. Thus, the promise of meeting a much-shortened schedule at a reduced cost cannot be reasonably considered without an examination of the processes that govern a nuclear plant delivery.

The development of a Process Model is then a summation of process insights from a number of other industries as well as retrospective reviews of what drove US nuclear construction costs high in the 1970's and 80's as well as what has been shown to be successful in foreign nuclear plant construction. Not withstanding an improved and less uncertain regulatory environment, the management of a nuclear plant delivery will benefit from application of the following technologies:

– Information Technology for shortened decision making and avoidance of rework in design activities;
– System Dynamics Modeling to model impact of changes both proposed and during project execution;
– Application of Baysien Belief Networks as online project management tools.

Process Model insights include:

- Use of a Central Data Warehouse to Avoid Design Rework;
- Data Warehousing needs to start on the first day of construction;
- Passive design tends to reduce safety related equipment count which reduces reliance on QA procurement and can simplify the supply chain despite an aggressive schedule;
- A combined model linking resources, schedule, purchased items and engineering assembly methods needs to be established;
- The previous custom of performing detailed engineering in parallel with major construction activities is not acceptable;
- Design and process planning must be comprehensive;
- 4D Modeling is essential to constructability;
- DPCIT process must become seamless such that unnecessary handoffs of information are avoided;
- All vendors must be part of the team to shorten decision making and action cycle;
- Once construction starts, there is no room for new technologies, therefore, new ideas need to be evaluated in prototype projects.

1.2.2. DPCIT summary

The nuclear industry recognizes the need to find methods to assure the next generation nuclear plants are created as timely and cost competitively as necessary to be a viable option for commercial electrical power generation. Both the utility industry funded 4D visualization task and the US government funded DPCIT task are discovering methods and tools that will assist in making the next generation nuclear powered electricity generation plant a viable option for supplying the world's electrical needs. By evaluating the complete nuclear plant creation process and developing tools that integrate and manage all the information require to complete the nuclear power plant project, the nuclear industry is well on its way to achieving it goal of competitive nuclear power plant design and construction.

2. SMART EQUIPMENT FOR FUTURE NUCLEAR POWER PLANTS (SMART-NPP) PROGRAM

The goal of the Smart-NPP program is to design, develop, and evaluate an integrated set of tools and methodologies that can improve the reliability and safety of advanced nuclear power plants through the introduction of "smart" equipment and predictive maintenance technology that ultimately aides in the reduction of construction, maintenance and operational costs. To accomplish this goal, the Smart-NPP program tasks are aimed at:

- Identifying and prioritizing nuclear plant equipment that would most likely benefit from adding smart features;
- Developing a methodology for systematically monitoring the health of individual equipment implemented with smart features (i.e. "smart" equipment);
- Developing a methodology to provide plant operators with real-time information through "smart" equipment Man-Machine Interfaces (MMI) to support their decision making;
- Demonstrating the methodology on a selected component; and
- Expanding the concept to system and plant levels that allow communication and integration of data among smart equipment.

2.1. Smart-NPP goals and significance

The subject of the Smart-NPP program inevitably raises the question: "What do you mean by Smart Equipment?" The answer is:

> *"Smart equipment embodies elemental components (e.g., sensors, data transmission devices, computer hardware and software, MMI devices) that continuously monitor the state of health of the equipment in terms of failure modes and remaining useful life, to predict degradation and potential failure and inform end-users of the need for maintenance or system-level operational adjustments."*

The results of the Smart-NPP program have the potential to substantially change the way that nuclear power plants are designed and operated. Nuclear power plant design today is often constrained by the need for frequent access to equipment for inspection and repair. Further, redundancy and diversity of equipment are needed to ensure safety and reliability under a variety of conditions. When combined with the NERI Risk-Informed program results[3], that move to a risk-based regulatory approach, the introduction of highly reliable 'smart' equipment and systems will allow plant designers to simplify plant designs without compromising reliability and safety. For example, normal operating systems employing smart components may supplement, or even replace, traditional safety systems such as Emergency Core Cooling or Emergency Feedwater. The smart features of the components may provide the basis for assuring that a non-safety system's availability is sufficient to meet Probabilistic Risk Assessment (PRA) goals and the demands of regulators. Such plant design innovation can potentially allow the use of less equipment resulting in more cost competitive and easier-to-construct power plants. Furthermore, the results of the Smart-NPP program will be useful to all reactor technologies (e.g., PWR, BWR, MHTGR, and PHWR), including new technologies that might be developed through other NERI projects (e.g., proliferation-resistant or low-output reactors).

A major contributor to high Operations and Maintenance (O&M) costs are maintenance practices that rely heavily on time consuming procedures. This includes periodic overhaul or replacement of parts is based primarily on historical maintenance records, without regard to the actual "health" of a component or system. The Smart-NPP results are providing a blueprint for creating the capability to predict system performance and remaining useful life with high confidence, based on predictive or condition-based maintenance methods that utilize current and projected conditions of critical components and subsystems to predict their time to failure. This requires understanding how an entire history or profile of sensor information, given specific environmental and operating conditions, relates to component or system wear and age. Such practices allow overhaul and repair to be performed only when necessary to prevent failure and provide a capability for assessing the risk of delaying indicated maintenance tasks. Maintenance methods that predict system performance while utilizing the maximum useful life of subsystems and components represent an innovative and cost saving approach to O&M activities. The overall reduction of the inventory of required plant safety equipment will likely produce an additional O&M benefit due to reduced surveillance testing requirements in Technical Specifications.

2.2. Smart-NPP accomplishments

The Smart-NPP team is presently embarking on the second year of its three year program with high expectations of realizing a demonstration health monitoring system tied to both a physical, real-world system and a virtual machine simulation by the year's end. The following is a listing of the significant achievements to date, which are further explained below.

- Developed system/component criteria to establish priorities for smart equipment application and used them to prioritize both PWR and BWR systems;
- Based on the prioritization, selected a high energy, horizontal, centrifugal pump as a demonstration component for a Health Monitoring System (HMS);
- Developed an architecture for a HMS using Bayesian Belief Networks (BBNs) to determine failure probability information based on sensor data and conditional probabilities;
- Procured the use of a pump lube oil system to supply real-world data to the HMS
- Created the design for a virtual machine for the selected pump to supply simulated reliability and sensor data to the HMS;
- Reviewed state-of-the-art pump diagnostics and assessed failure modes of the pump to provide the basis for establishing an optimum health monitoring plan;
- Reviewed and assessed sensor technology to develop criteria for sensor element selection and sensor system architecture;
- Reviewed smart equipment MMI technology currently being used in other industries to support creation of an MMI prototype;
- Established industry contacts for potential cooperative working arrangements.

2.2.1. Task 1: System evaluation and prioritization study

This initial Smart-NPP task has been completed during the first project year. The results are (1) a methodology for systematically evaluating plant structures, systems and components (SSCs) to determine those that would benefit most from application of smart equipment concepts, (2) selection of a demonstration component and (3) an optimum health monitoring plan for the selected component, including identification of its failure modes.

A study of failure rates and failure modes considered data of SSC contributions to forced outages. This study used the NRC MORP 2 Database for Monthly Reports between 1990 and 1999 for 14 PWR and 13 BWR units. SSCs were ranked based on their fraction of the total forced outage time (based on occurrence frequency and mean outage duration). Individual failure modes were similarly ranked for the SSCs with the highest forced outage contributions. This quantitative data was combined with qualitative team assessments of instrumentation feasibility and cost/benefit to result in a SSC prioritization. The significant result of this effort was identification of rotating machinery, including pumps, as the primary contributors to forced outages in LWRs. This conclusion, coupled with their application in both charging and feedwater systems, led to the selection of a high energy, horizontal, centrifugal pump as the demonstration component for the Smart-NPP project.

The other Task 1 effort explored the nuclear industry's transition from traditional time-based and corrective maintenance methods to Reliability Centered Maintenance (RCM), including application of Condition Based Maintenance (CBM). Methods for monitoring component health being developed in the Smart-NPP program directly support the transition to CBM. Typical pump failure modes were identified and are described fully in [4]. Current pump diagnostics however are often limited to characterizing casing vibration via portable sensors. Integration of advanced diagnostic methods including vibration analysis, rotor dynamics modeling, infrared thermography, motor monitoring, lubrication assessment, acoustic monitoring and performance parameter measurement will be critical to developing an optimum HMS for a pump. Other issues identified as critical to the effectiveness of an HMS include (1) sensor adequacy and location, including potential use of "smart" sensors, (2) data acquisition, particularly with respect to assessing the benefits offered by wireless data transmission and (3) selection of algorithms and intelligent processing systems to process the data into useable information. The full results of the optimum HMS evaluation are provided in [4].

2.2.2. Task 2: sensor technology and installation analyses

Task 2 has featured three somewhat independent aspects of smart equipment development during the first project year. These are (1) sensor selection criteria, (2) use of plant system modeling to support sensor development and (3) a technology assessment of MMI techniques being employed in smart equipment applications in other industries.

Criteria for sensor selection have been developed for both sensor elements and sensor system architectures. Key criteria identified pertaining to sensor elements are (1) the ability to indicate component state based on either the physics of failure mechanisms or a Failure Modes and Effects Analysis (FEMA), (2) the ability to withstand the local environment (e.g., temperature or radiation effects), (3) accuracy and (4) reliability. The criteria identified for a sensor system architecture include (1) flexibility, (2) a web-based design including compatibility with the IEEE 1451 standard, and (3) a wireless data communications network. Of particular note is the potential for wireless data communications to minimize concerns associated with installation feasibility and the cost of wired communication networks. Based on current industry direction, it is recommended that smart equipment networks be compatible with the “Bluetooth” wireless protocol, which is emerging as an industrial standard.

For high-energy pumps, diagnostic technology in today’s nuclear plants is quite out-of-date. Rotor/bearing dynamic modeling has proven effective in extending the effectiveness of a limited number of sensors in today's pumps. To support development of smart equipment, the failure modes identified in Task 1 were addressed via rotor/bearing dynamics modeling. This effort is resulting in recommended enhancements in sensor placement and sensor development. Additionally, dynamic modeling is being calibrated with pump operating data to provide an array of “virtual” sensors that can aggressively assess the condition of equipment and supply input data to the HMS BBNs. An effort is underway to determine how to best integrate the pump dynamic modeling with the virtual machine pump model.

The MMI technology assessment investigated smart equipment applications in other industries for potential use in nuclear power plants. The technology assessment identified various techniques for presentation of smart equipment and predictive maintenance information, including display and warning techniques. An example of smart equipment MMI is provided in Figure 5. Another result of the investigation was the potential use of smart equipment in control applications. The aerospace industry uses agents to both sense and control a dynamic environment to accomplish a predetermined goal. This has the potential in future nuclear plants to move smart equipment from the realm of only monitoring to that of automatic control.

In the next project year the sensor technology and installation task focuses on evaluating advanced sensor technology for applications supporting smart equipment use in nuclear power plants. A methodology for performing sensor installation feasibility studies will be developed and applied to the horizontal, centrifugal pump. Future MMI work will concentrate on developing a smart equipment display set and display features with the end result being a prototype display set for the pump demonstration facility. A human factors validation will assess usability of this MMI from both an operations and a maintenance perspective.

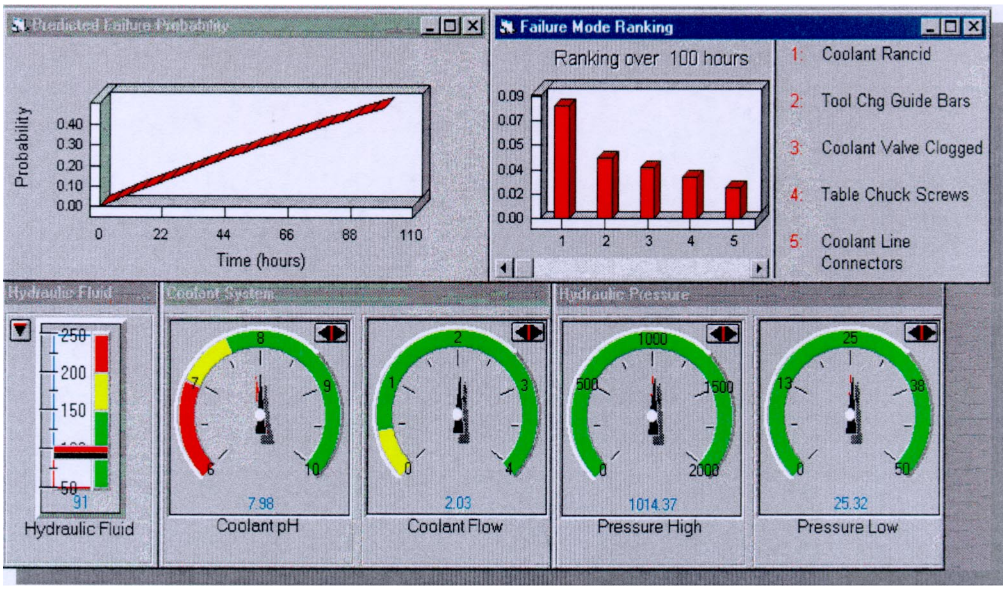

FIG. 5. Example man-machine interface for a smart component.

2.2.3. Task 3: Equipment maintenance and reliability simulation ("Virtual Machine") capability

The efforts of Task 3 are developing a virtual machine for the centrifugal pump with the capability to simulate equipment behavior, such as failures, maintenance (including inspection and repair activities) and user-defined sensor signals. The virtual machine supports design and testing of the HMS, allows evaluation of the benefits of incorporating smart features and provides a platform for realistic demonstrations. Figure 6 illustrates the overall architecture of a HMS with a virtual machine simulating an actual plant component.

The virtual machine depicted in Figure 6 consists of three primary components: a reliability module, a scheduling module and a simulation engine. The reliability model identifies failure modes and their relationships including maintenance impact and effects of aging, based on historical data supplemented with engineering judgement. The scheduling module defines schedules for equipment use and maintenance. The simulation engine generates the components behavior (e.g. state changes) based on inputs from the scheduling module and reliability model and provides it as input to the Computerized Maintenance Management System (CMMS) and the HMS software.

2.2.4. Task 4: Smart equipment health monitoring system

Developing methods for taking sensor data from the component monitoring and translating it into information relative to the equipment's health is the heart of the Smart-NPP program. Equipment health can include information about predicted lifetime of the equipment, estimated percentage wear out on various components, recommendations for preventive maintenance activities, predictions of likely failure modes and causes and cost impact of maintenance-related decisions.

A significant accomplishment early in the first project year was the decision to follow the smart equipment methodology outlined in [5]. This previous work at MIT provides a structure for developing comprehensive sensor networks and analysis of the resultant data to create an intelligent diagnostic and maintenance advisory system. Adoption of this methodology has provided direction for development of the demonstration HMS. Specifically fault trees have been constructed providing a functional decomposition of the centrifugal pump. Starting at the highest level of "pump failure" the fault trees break down pump subsystems until individual cause-consequence branches are identified.

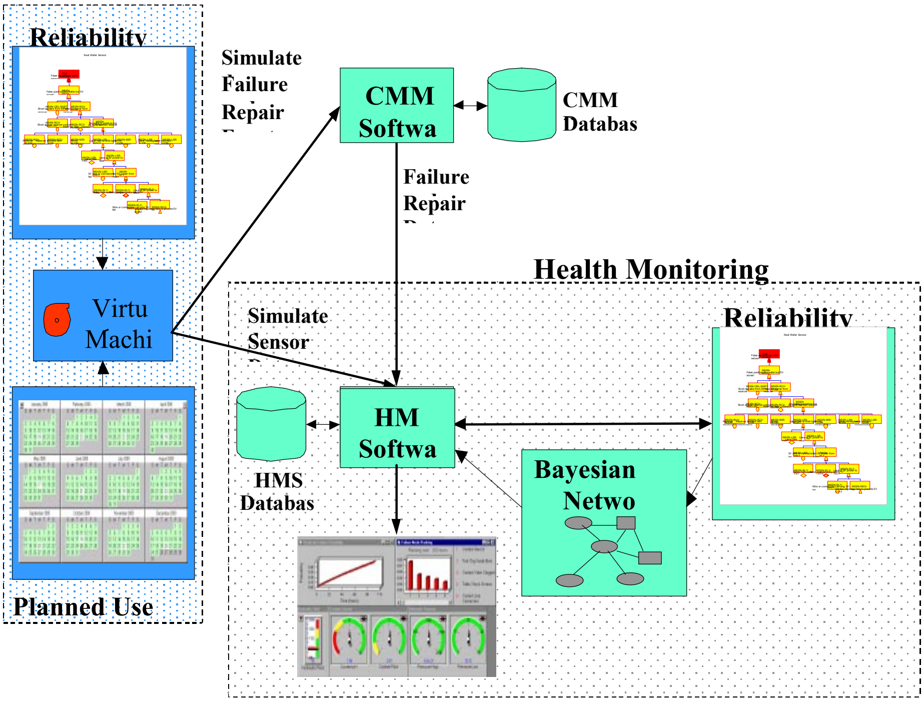

FIG. 6. Health monitoring system linked to a virtual machine.

Also of importance to the HMS development is the endorsement of Bayesian Belief Networks (BBNs) as the engine needed to capture the expertise relating sensor data to system states through the use of conditional probabilities. The BBN approach was selected because (1) it has been shown to work better than rule-based and neural network systems, (2) it is very flexible and tolerant of complexity and (3) it is available on personal computer with a convenient user interface [5]. The “HUGIN” BBN shell has been selected for use on the project and an initial canned demonstration of its application has been completed. The effort to populate the conditional probabilities based on input from pump and maintenance experts has been initiated. Development and population of the BBNs for the centrifugal pump will continue throughout the next project year.

2.2.5. Task 5: Sample application of health monitoring system

Perhaps the most significant accomplishment of the Smart-NPP program to date is the selection of high energy, horizontal, centrifugal pumps as a demonstration component. This pump is used in both charging and feedwater systems for PWRs and was selected based on the criteria established in Task 1. Its selection has allowed subsequent program activities, such as the virtual machine design, to focus methodological developments on a specific application.

Another important milestone has been the identification of a related test bed. The Smart-NPP team concluded that a software only demonstration using the virtual machine could be perceived as doing little to address real world problems in developing a HMS. For example, data acquisition may be much more difficult from an actual sensor network, compared with simulated sensor data acquisition. To address this concern, a pump lube oil test system at Penn State University (see Figure 7) will be utilized for instrumentation and testing of an actual subsystem typical of the selected centrifugal pump. The virtual machine will simulate the remainder of the pump to allow testing of a HMS for the entire component as described in Task 3. A basic structure of the integrated demonstration system is shown in Figure 8. The current goal is to make this a web interface to allow testing and demonstration of the HMS at a variety of locations.

The eventual HMS demonstration will help develop the methodology for systematically evaluating equipment to determine how best to improve its reliability. In addition, it will provide an opportunity to evaluate and optimize 'smart' equipment and predictive maintenance strategies and support the MMI validation.

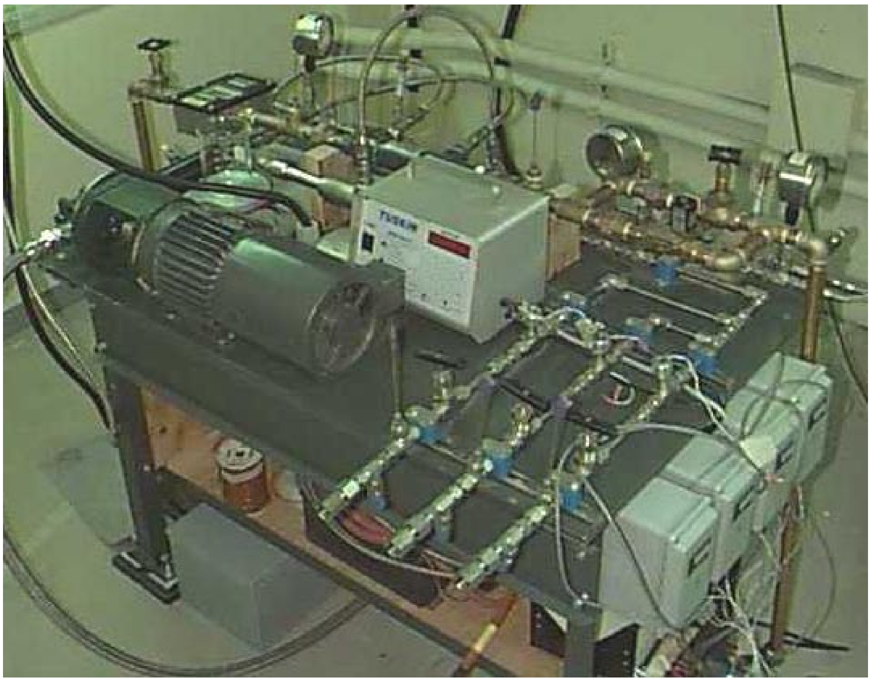

FIG. 7. Pump lube oil system at Penn State University.

2.2.6. Task 6: Enterprise level health monitoring

This task will develop a methodology that combines equipment-health information from individual components into overall plant-health information. It will expand the health-monitoring concept to system and plant levels, allowing communication and integration of data among the smart equipment, as well as control room systems and plant operators. An advanced information system architecture will be designed to support data transfer and storage at the enterprise scale.

The system will be designed to:

- Provide data and configuration information required for interpreting and displaying real-time sensor and health data at the component, system, and plant levels;
- Provide historical performance and maintenance data required for analyzing reliability, spares, and maintenance conditions;
- Store component, system, and plant configuration models and simulation data;
- Support data requirements of selected reliability and maintenance analysis techniques.

2.3. Smart-NPP summary

The results of the Smart-NPP program have the potential to substantially change the way that future nuclear power plants are designed and operated. By providing the capability to predict future component and system performance with high confidence, the development of smart equipment will help improve the cost competitiveness of nuclear power by (1) providing substantial operations and maintenance savings and (2) reducing capital costs by allowing front-line systems in normal operation to supplement or even replace dedicated safety systems.

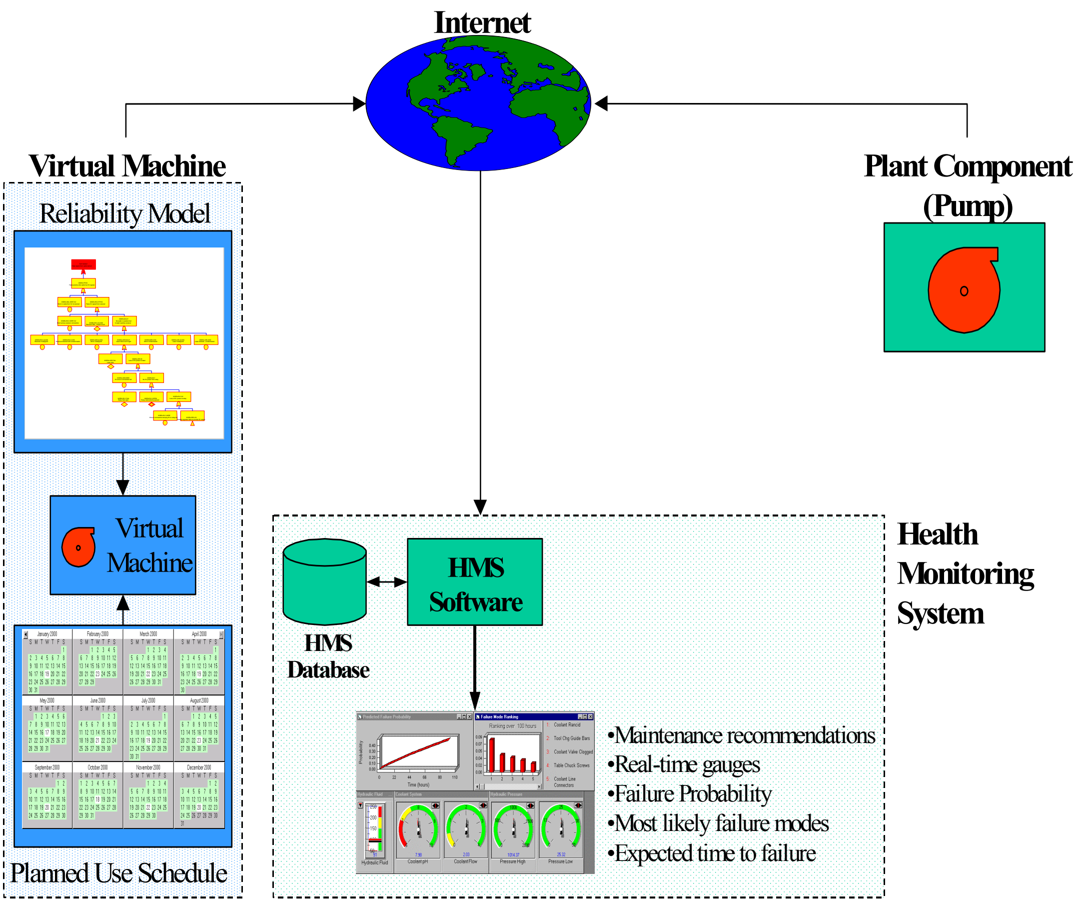

FIG. 8. HMS linked to a virtual machine and physical system.

Upon completion of its first year, the Smart-NPP program is well on its way to achieving the program's goal of designing, developing and evaluating a health monitoring system for a nuclear plant component. Significant achievements this year include:

– Selecting a high energy, horizontal, centrifugal pump, based on SSC prioritization criteria, as a demonstration component for a HMS;
– Developing a HMS architecture using Bayesian Belief Networks to relate sensor data to failure probability;
– Creating a combination of real-world and simulated input data for the HMS through use of a pump lube oil system and creation of a virtual machine, respectively;
– Reviewing and assessing sensor and smart equipment MMI technology as precursors to creating the demonstration system;
– Establishing industry contacts for potential cooperative working arrangements.

The Smart-NPP team is continuing to make progress, with an eye toward making the best use of industry and international cooperation to extend the potential results of the program.

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