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FOREWORD

To support the IAEA's function of encouraging the development of atomic energy for peaceful uses throughout the world, the IAEA programme in nuclear power technology development promotes technical information exchange and co-operation between Member States with major reactor development programmes, offers assistance to Member States with an interest in exploratory or research programmes, and publishes reports which are available to all Member States interested in the current status of reactor development. The IAEA's activities in the development of water cooled, liquid metal cooled and gas cooled reactors are co-ordinated by three International Working Groups (IWGs) which consist of representatives of national programmes and international organizations in these technologies. Each IWG meets periodically to serve as a global forum for information exchange and progress reports on national programmes, to identify areas of common interest for collaboration and to advise the IAEA on its technical programmes and activities.

Water cooled reactors represent, both in number and capacity, a clear majority in the current fleet of operating nuclear power plants throughout the world. Therefore, the further development of water cooled reactors is, and has been for quite some time, a matter of interest for the IAEA, and survey reports reflecting the status of development in the Member States have been published periodically. In the field of light water cooled and moderated reactors (LWRs), the latest report, Status of Advanced Technology and Design for Water Cooled Reactors; Light Water Reactors (IAEA-TECDOC-479), was published in 1988. Since then quite a lot has happened, however; some designs have become commercial products, for others significant steps have been taken towards commercial deployment, while others have begun their development after 1988. Therefore, the International Working Group on Advanced Technologies for Water Cooled Reactors (IWGATWR) has recommended the preparation of a new status report on advanced LWR designs.

At the Technical Committee Meeting on Review of Design Approaches of Advanced Pressurized LWRs in Moscow in 1994, (reported in IAEA-TECDOC-861), it became evident that there had been significant technological advances as well as changes in the economic and regulatory environment since the 1988 report; a new report was really needed. Therefore, the AIEA intensified its data acquisition efforts for the report and called upon nuclear vendors to provide descriptions.

The present report, which is significantly more comprehensive than the previous one, addresses the rationale and basic motivations that lead to a continuing development of nuclear technology, provides an overview of the world wide status of current LWRs, describes the present market situation, and identifies desired characteristics for future plants. The report also provides a detailed description of utility requirements that largely govern today's nuclear development efforts, the situation with regard to enhanced safety objectives, a country wise description of development activities, and a technical description of the various reactor designs in aa consistent format. The reactor designs are presented in two categories: (1) evolutionary concepts that are expected to be commercially available soon, and (2) innovative designs. The report addresses the main technical characteristics of each concept without assessing or evaluating them from a particular point of view (e.g. safety or economics). Additionally, the report identifies basic reference documents that can provide further information for detailed evaluations. The report closes with an outlook on future energy policy developments.

In preparing the report, the IAEA staff has supplemented its available information with data and descriptions provided by suppliers for each reactor concept under development. Country wise information on national energy policies, national LWR development programmes including research and performance of existing nuclear capacity, up through December 1995, were provided by IWGATWR members. Finally, an Advisory Group meeting held in Vienna in October, 1996 reviewed the draft and provided additional information for integration into the report.

The report is intended to be a source of objective reference information for interested organizations and individuals. Among them come, in first line, decision makers of developing countries intending to implement nuclear power programmes. Further, the report is addressed to governmental officials with technical background and to research institutions of countries with existing nuclear programmes, which wish to evaluate alternatives for future energy policies, as well as to members of the public that look for unbiased information on the nuclear issue.

The IAEA is grateful to the many experts who have contributed to this publication, either by providing information and material or as participants at the meetings. The responsible officer of the IAEA for the TECDOC was T. Pedersen of the Division of Nuclear Power and the Fuel Cycle.

EDITORIAL NOTE

In preparing this publication for press, staff of the IAEA have made up the pages from the original manuscripts as submitted be the authors. The views expressed do not necessarily reflect those of the IAEA, the governments of the nominating Member States or the nominating organizations

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CONTENTS

CHAPTER 1. TRENDS IN ADVANCED LWR DESIGN AND TECHNOLOGY 9

1.1. Incentives for the development of advanced LWRs 9

1.1.1. Prospects for increased nuclear power deployment 9
1.1.2. Current status of nuclear power utilization 10
1.1.3. Directions for development of advanced LWRs 11
1.1.3.1. Technology development 11
1.1.3.2. Standardization efforts 13

1.2. Design objectives for advanced LWRs 15

1.2.1. Market situation 15
1.2.2. Improvement needs 15
1.2.3. Desired characteristics of advanced designs 16
1.2.4. Other acceptability considerations 17

1.3. Outline of the development of advanced LWRs 19

1.3.1. A general overview of advanced design development 19
1.3.2. Overview of advanced light water reactor programmes 21

CHAPTER 2. UTILITY REQUIREMENTS AND SAFETY OBJECTIVES FOR ADVANCED WATER COOLED REACTORS 25

2.1. Development of utility requirements documents (URD) 25

2.1.1. EPRI Utility Requirements Document (URD) 25
2.1.2. European Utility Requirements Document (EUR) 26
2.1.3. Japanese Utility Requirements Document (JURD) 27
2.1.4. Korean Utility Requirements Document (KURD) 28
2.1.5. Taiwan Power Company Requirements Document (TPCRD) 29

2.2. Top level utility requirements 29

2.2.1. General overview 29
2.2.2. EPRI ALWR URD requirements 29
2.2.3. EUR requirements 31
2.2.4. Japanese requirements 33
2.2.5. Commonalties and differences 35

2.3. Safety objectives 36

2.3.1. Current licensing and design basis requirements 36
2.3.2. Expanded safety strategy 36
2.3.3. Extended design conditions 37
2.3.4. Identification of severe accidents to be addressed 38
2.3.5. Severe accident challenges 39

CHAPTER 3. ALWR DEVELOPMENT ACTIVITIES 41

3.1. Countries with programmes and activities related to advanced nuclear power plant projects 41

3.1.1. Finland 41
3.1.2. France 44
3.1.3. Germany 47
3.1.4. Japan 51
3.1.5. People's Republic of China 54

3.1.6. Republic of Korea 56
3.1.7. Russian Federation 59
3.1.8. Sweden 62
3.1.9. United Kingdom 66
3.1.10. United States of America 68

3.2. Countries with programmes and activities related to specific design features of advanced nuclear power plant projects 72

3.2.1. Belgium 72
3.2.2. Italy 76
3.2.3. Netherlands 80
3.2.4. Spain 83
3.2.5. Switzerland 86
3.2.6. The Czech Republic 89

CHAPTER 4. EVOLUTIONARY, LARGE SIZE ALWR DESIGNS (850 MWe or larger) 91

4.1. ABWR, GE, USA in co-operation with Hitachi and Toshiba, Japan 91
4.2. APWR, Mitsubishi, Japan/Westinghouse, USA 119
4.3. BWR 90, ABB Atom, Sweden 139
4.4. EP 1000, Westinghouse, USA/Genesi, Italy/EUR 163
4.5. EPR, NPI, France/Germany 190
4.6. ESBWR, GE, USA 218
4.7. Korea Next Generation Reactor (KNGR), KEPCO, etc., Republic of Korea 242
4.8. Sizewell C, NNC, UK 265
4.9. SWR 1000, Siemens, Germany 291
4.10. System 80+, ABB CENP, USA 313
4.11. WWER-1000 (V-392), Atomenergoproject/Gidropress, Russian Federation 343

CHAPTER 5. EVOLUTIONARY, MEDIUM SIZE ALWR DESIGNS (the 500-800 MWe range) 361

5.1. AC-600, China National Nuclear Corporation, China 361
5.2. AP-600, Westinghouse 377
5.3. HSBWR, Hitachi, Japan 406
5.4. MS-600, Mitsubishi, Japan 417
5.5. SBWR, GE 438
5.6. WWER-640 (V-407), Atomenergoproject/Gidropress, Russian Federation 440

CHAPTER 6. DESIGN CONCEPTS REQUIRING SUBSTANTIAL DEVELOPMENT 459

6.1. ISIS, Ansaldo, Italy 459
6.2. JPSR, JAERI/IHI, Japan 476
6.3. PIUS, ABB Atom, Sweden 491
6.4. SPWR, JAERI/IHI, Japan 513
6.5. VPBER-600, OKBM, Russian Federation 529

CHAPTER 7. OUTLOOK AND EXPECTATIONS FOR THE FUTURE 549
7.1. Nuclear power projections to 2015 549
7.2. Non-electric applications of nuclear energy 550
7.3. Nuclear power development in industrialized countries 551
7.4. Nuclear power development and developing countries 553

Annex: NUCLEAR POWER DEVELOPMENTS IN THE WORLD 555

RELATED IAEA PUBLICATIONS 573

CONTRIBUTORS TO DRAFTING AND REVIEW 575

CHAPTER 1. TRENDS IN ADVANCED LWR DESIGN AND TECHNOLOGY

1.1 INCENTIVES FOR THE DEVELOPMENT OF ADVANCED LWRs

1.1.1 Prospects for increased nuclear power deployment

Global climate change is considered to be a major environmental threat, and greenhouse gas emissions from electricity generation chains are the focus of attention of energy planners and decision makers at the national and utility levels. Recent studies on greenhouse gas emissions from different energy chains for electricity generation indicate that nuclear power is one of the better options to alleviate global climate change (Figure 1-1). Renewable sources emit in the range of 2 to 5 times more greenhouse gases than nuclear power per unit of electricity produced when the entire fuel chain is considered, and the emissions from fossil fuel chains range from 40 to 100 times more!

However, nuclear power alone will not ensure a secure and sustainable electricity supply world wide, nor will it be the only means of reducing greenhouse gas emissions, but it has a key role to play in this regard. A prerequisite for realizing the necessary revival of the nuclear option is that the technical and economic performance of nuclear power plants must improve while maintaining a high safety level and satisfactorily addressing the issues of waste management and disposal.

Nuclear power is among the cheapest sources of electricity generation in many countries, although its competitive margin has been reduced by low fossil fuel prices, improved efficiencies and increases in nuclear power plant capital costs; the latter very much due to long construction and licensing lead times for current generation of nuclear power plants. The economic advantage of nuclear power can be maintained or re-inforced by the efforts of reactor designers to reduce capital costs by standardizing the reactor concepts, design simplification, and optimization of the construction times.

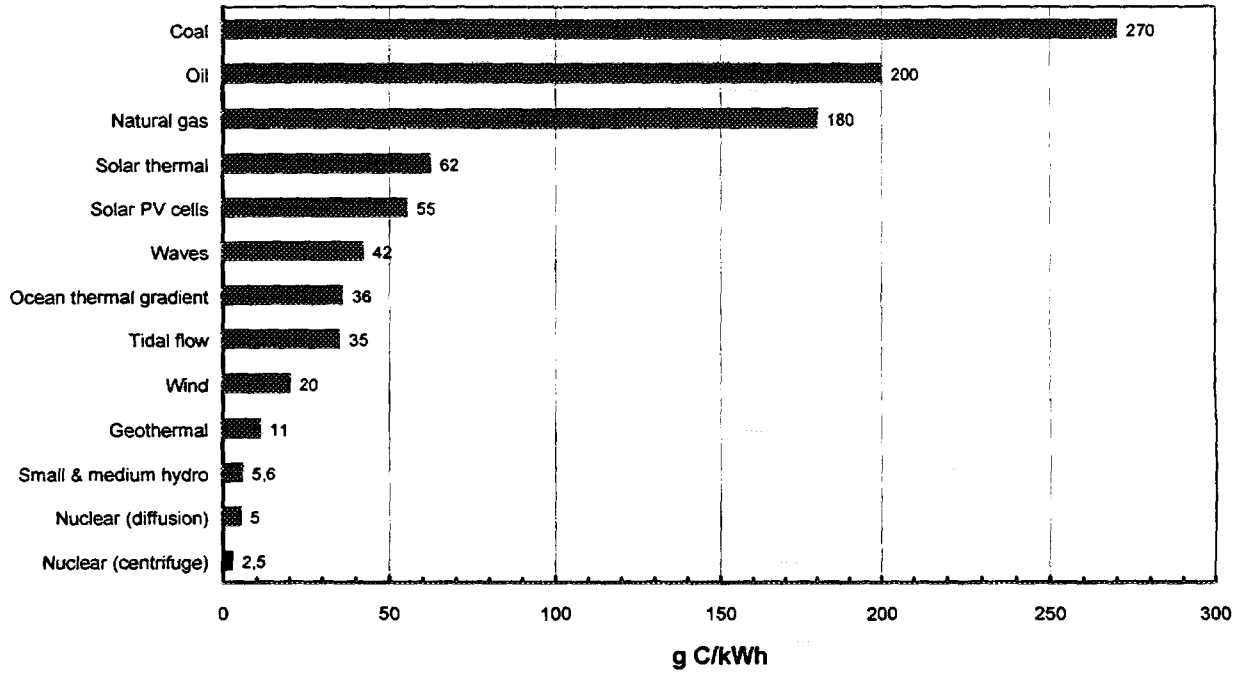


Figure 1-1 CO_2 emission factors from the full energy chain of different power sources [Source: "Atoms in Japan", November 1994]

Substantial progress has been achieved in this regard and additional gains are expected through the deployment of advanced reactors which are being developed for all principal reactor types. Furthermore, rising market prices of fossil fuels, in particular gas, and increased capital and operating costs of fossil fired units due to the required addition of abatement systems may reverse this trend in the coming decade.

Financing the large capital costs of nuclear power plants will remain a key issue in developing countries. Technology adaptation, the development of small and medium size reactors, and the implementation of new financing models, e.g., by establishment of international funds, may alleviate the funding constraints and facilitate a broader deployment of nuclear power in developing countries.

1.1.2 Current status of nuclear power utilization

Nuclear power plants that utilize water cooled reactor technologies are clearly dominating among the operating nuclear power plants throughout the world, both in number and total power; according to the IAEA PRIS data base a total of 396 of the 437 units in operation at the end of 1995 were of water cooled type, with a capacity of 330.1 GWe out of a total worldwide nuclear power capacity of 344.4 GWe. In other words, of all power reactors worldwide, water cooled reactors are roughly 90% in number and 96% in power.

The water cooled reactor technologies cover a wide range of different design lines; there are light and heavy water cooled reactors with various types of moderator. The clearly dominating types are the light water cooled and moderated reactors (LWRs). Today, 343 LWRs are in operation, representing 78% of the total number of nuclear power plants, with a capacity of 297.1 GWe - roughly 86% of the total nuclear power capacity. There are 33 heavy water cooled and moderated reactors (HWRs) with a capacity of 18.0 GWe, 20 light water cooled and graphite moderated reactors (LWGRs or RBMKs) with a power of about 15 GWe, and one light water cooled, heavy water moderated reactor of 148 MWe, making up the remainder of the water cooled reactor group. LWRs are in turn divided into boiling water reactors (BWRs) and pressurized water reactors (PWRs). The latter account for 250 out of the total 343 LWRs in operation, or close to 60% of the total number of nuclear power plants worldwide, and more than 50% of the total nuclear capacity.

Nuclear power has proven its viability as an energy source in many countries. The nuclear technology is mature and well proven, and it has had tremendous progress and consolidation in the last decades. A very broad experience of operating nuclear power plants is available; in the United States of America more than 100 nuclear power plant units are in operation with a total experience base of more than 1800 reactor years, and the total operating experience worldwide amounts to well above 7700 reactor years.

The current LWR and HWR technologies have proven to be economic, safe and reliable, and they have a mature infrastructure and regulatory base in several countries. The accumulated operating experience of the three dominating reactor types amounts to about 3300 reactor years for the PWRs, 1400 reactor years for the BWRs, and 450 reactor years for the HWRs, providing a significant experience basis for further product improvements.

The development of nuclear energy has attained a very high standard in reliability and availability, and a very high level in performance and safety. Plant availabilities above 80% are more and more becoming normal, rather than exceptional achievements. Load following operations with LWR plants have been successfully demonstrated and their load following capability is fully compatible with that of conventional fossil-fired power plants. The electricity generation cost of nuclear power plants is comparable with that of coal-fired plants; in some regions it is even considerably cheaper. Construction activities are now controlled in a better way; by adoption of adequate project management, scheduling and licensing construction times of 5-6 years could be met. Overall, a high quality of operation and maintenance has been attained, in compliance with strict safety requirements, incorporating feedback and lessons learned from operating experience.

Many countries are heavily dependent on nuclear power for their electricity production; Belgium, Bulgaria, Finland, France, Germany, Hungary, Japan, Lithuania, the Republic of Korea, Slovenia, Spain, Sweden, Switzerland and Ukraine are examples on countries where nuclear power accounted for some 30% or more of the electricity generation in 1995 (Figure 1-2).

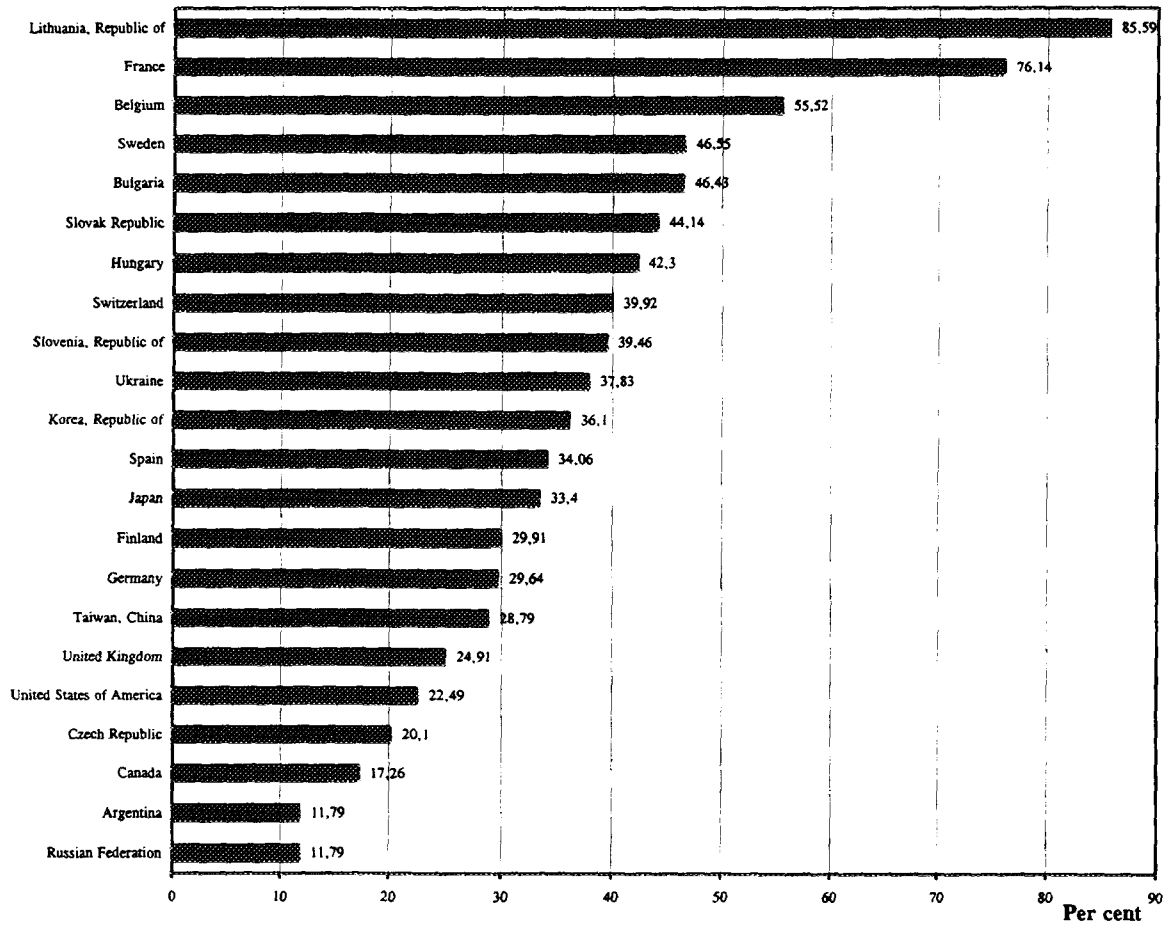


Figure 1-2 Nuclear power share of electricity generation in 1995 (in countries with more than 10% nuclear share)

The construction of new power plant capacity, nuclear as well as fossil-fueled plants, has slowed down, since the growth in demand for electrical energy in industrialized countries has been low for many years. Nuclear power plant construction involves high investment costs, and the long delays for completion of nuclear power plant projects that have been experienced in a number of countries have made utilities reluctant to invest in new nuclear capacity. Furthermore, public concerns about nuclear safety and the potential for severe accidents with significant releases of radioactive material to the environment have resulted in a complete re-examination of the nuclear option in a number of countries. As a consequence, the number of nuclear power plants taken into operation each year is rather small; national programmes for new nuclear capacity exist only in a few countries, such as, France, India, Japan, the Republic of Korea, Russia, and the Peoples Republic of China, as well as Taiwan, China. Still, even though the high financial burden incurred with nuclear power plant construction make utilities reluctant to select the nuclear option, the worldwide installed nuclear capacity for electricity generation could, according to the forecasts of the World Energy Council (WEC), be expected to increase by 10-15% over the next decade, due to growing concerns about the environmental effects of burning fossil fuels for energy production; nuclear energy represents a very benign energy source from the environment point of view.

Substantial design and development programmes are underway in a number of Member States for further technology improvements and for development of new reactor designs. As noted above, the major portion of existing nuclear power plants consists of light water reactor plants, and it is envisaged that among all the different lines of nuclear reactor types, the main stream of nuclear power deployment in the world in the next decade(s) will consist of LWR plants.

1.1.3 Directions for development of advanced LWRs

1.1.3.1 Technology development

As mentioned above, LWR plants offer a broadly developed and mature technology basis, but there is still a potential for further improvement; history shows that technology advances and entirely new knowledge, as well

as operating experience, provide the basis for continual stepwise improvements of plant or equipment designs in all areas of industrial activity.

Therefore, with time nuclear power plant designs will inevitably change, to adapt to new generations of equipment and to implement operating experience, but designers and utilities will also have an ambition of making the best possible product. Different advanced designs[1] a wide range of different objectives and goals; an emphasis on economy, plant reliability, availability, and safety represents a common theme for most of the new designs, however.

From a technical point of view, this term advanced design does not, per se, address specific improvements of plant characteristics, or particular enhancements with respect to design features and performances; it merely implies that the "new" designs differ from its predecessors or from existing designs, and are expected to bring improvements of some kind.

The development activities related to advanced plant designs can also be divided up into two separated lines, based on the degree of deviation from existing designs, or rather the need for confirmation and validation before commercial deployment. The first of these categories is called "evolutionary designs"[2]; basically, these represent descendants from existing plant designs featuring improvements and modifications based on feedback of operating experience and/or aimed to adopt new technological achievements. The other category covers design concepts that will incorporate more significant departures from existing nuclear power plant designs, and will require much more development effort, possibly including construction and operation of a prototype and/or demonstration plant before their commercial deployment. This category does not have any established name ("developmental designs" has been used tentatively by some organizations) and is best described by the attribute "design concepts requiring substantial development[3].

Some of the evolutionary designs build closely on earlier designs and need only engineering efforts before a commercial deployment; the deviations from existing designs are limited and the financial uncertainty for a utility will be rather small. Other evolutionary designs will incorporate more new features and depart more from existing designs, e.g., by introduction of passive safety features to replace, or supplement, traditional safety features and systems to achieve plant simplification. These designs will generally, in addition to the engineering efforts, need also confirmatory testing of the new features and components prior to the commercial deployment.

The range of designs for which substantial development efforts are still needed is much wider than for the category of evolutionary designs. For some concepts, the development may have been under way for many years and may be almost completed, while for others the activities have been initiated more recently and much work remains to be performed. It can also be observed that while some of the concepts have been under development for decade(s), there are others that represent new approaches. A sub-category of such designs is the innovative designs[4] ;or also novel designs, that designate designs that incorporate radical conceptual changes in design approaches or system configuration in comparison with established practice.

Since the degree of innovation introduced may differ from concept to concept and since the judgement of which changes are radical conceptual changes is necessarily very subjective, a sharp and objective distinction between innovative designs and other designs needing substantial development is very difficult. An explanation of the terminology is therefore necessary when describing reactor concepts of this kind.

LWR technology improvements under development encompass several different directions; some aim at better fuel utilization in existing light water cooled reactors, by increasing fuel discharge burn-up and better

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[1] An advanced design is a proposed design of current interest or merit that has not yet been constructed or operated (from IAEA-TECDOC-936, Terms for Describing New, Advanced NPPs).

[2] An evolutionary design is an advanced design which requires at most engineering or confirmation testing before a commercial development (from IAEA-TECDOC-936, Terms for Describing New, Advanced NPPs).

[3] A design concept requiring substantial development is an advanced design that requires, or may require, a prototype or a demonstration plant before a commercial deployment (from IAEA-TECDOC-936, Terms for Describing New, Advanced NPPs).

[4] An innovative design is an advanced design which incorporated radical conceptual changes in design approaches or system configuration in comparison with established practice. Substantial R&D, feasibility tests, and a prototype or demonstration plant are probably required (from IAEA-TECDOC-936, Terms for Describing New, Advanced NPPs).

optimization of core designs; including utilization of advanced burnable absorber strategies, increasing water inventory in fuel assemblies to improve neutron economy, and adding low-power boundary regions with natural uranium, whereas others aim at wider improvements of specific or overall responses of new designs to disturbances, transients and accidents.

Management of the radioactive waste that arises from the operation of nuclear power plants is an issue of major concern, for the utility, the regulators, the politicians, and the public. The handling of the most radiotoxic waste, the spent fuel, follows two general, main routes. In the first, the currently preferred alternative in some countries, e.g., France and Switzerland, the spent fuel is treated in a chemical process that separates plutonium and other fissile material, which is then utilized for manufacturing of mixed oxide (MOX) fuel, and then "recycled". In the other, the currently preferred alternative in Finland and Sweden, the spent fuel will be stored in water pools for some 30 years, after which it will be "packed" into canisters and transported to a final rock repository located several hundred meters below grade. The proponents of the first line refer to the ecological principles; "useful" material in waste shall be retrieved as far as possible and recycled, and the remaining waste shall be handled in a safe manner. The proponents of the other line oppose the arguments for the first line, and conclude that the reprocessing process is much too expensive; the fuel value of the retrieved material is too low to compete with new uranium at current prices and therefore, the direct disposal approach is by far the most economical.

In the short to mid-term time perspective, another problem related to supply/disposal of fissile material appears to be important. A significant amount of weapons-grade material has been produced around the world for use in deterrent nuclear weapons. Now, this material is no longer needed, and ways are sought to dispose of it, together with material from dismantled nuclear weapons, in a controlled and safe manner. In this context, the main alternatives are disposal or utilization as fissile material in nuclear power plants.

So-called high conversion reactors in which a tighter core lattice is introduced have been seen as a possible way of reducing the uranium feed needs. However, the studies are no longer pursued intensively, partly as a result of the low market prices for uranium, and partly due to the gloomy near term prospects for new nuclear power plant construction. The high conversion reactors represent a development project with additional investment costs since these reactors require changes of the core, its internals and the control rods; besides, new issues will have to be dealt with in the licensing process.

1.1.3.2 Standardization efforts

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

With respect to the latter, the efforts by the utilities, and by the regulators, are very important, and in this context, the Utility Requirements Document (URD) that has been developed by the Electric Power Research Institute (EPRI), should be highlighted. Later on, utility groups in other regions of the world have developed, or are developing, their own requirements, e.g., the European Utility Requirements (EUR) document, following similar lines. Even though these activities start as requirements for a limited geographical regions, attempts are currently underway to harmonize them as far as reasonably achievable - to establish a "world standard" to be used by all utilities when calling for bids. The different vendors/ designers of nuclear power plants will undoubtedly try to adapt their products to such requirements.

The relationship between utilities-vendors and regulators has changed in a positive way; utilities and regulators have opened up a dialog regarding safety goals and ways to achieve them, and contacts between vendors and regulators, already in the early stages of the design work, have also been found beneficial - and acceptable. In addition, nuclear regulators in different countries, e.g. in France and Germany, have started co-

operation to try to harmonize their requirements, and this will hopefully result in the establishment of an international basic safety standard to be met by all new nuclear power plants.

The utility requirements documents, in particular the URD, comprise requirements at different levels; partly general objectives or functional requirements, so-called top-tier requirements, and more detailed requirements on system levels. The latter are generally developed for a particular design and normally very specific for that design, and therefore they will have a limited applicability, whereas the top-tier requirements will be the general yardstick for any new design to be proposed.

The general requirements ask for increased margins, design simplifications, reduced dependence on operator actions, increased error tolerance, and improved reliability, increased design lifetime, reduced construction periods, shortened refuelling outages, reduced occupational exposures, and longer fuel cycle lengths, define specific goals for reduced core damage frequency and probabilities of large releases to the environment, including consideration of severe accidents, stipulate limits for acceptable specific installation cost (cost per kW) and for the energy cost over the lifetime of the plant, and, last but not least, demand that new plants must be perceived as a "good neighbour". In summary, they require that new plants be safe, user-friendly and economic.

Some of the new, advanced reactor designs under development are being developed "in parallel" to the evolvement of the requirements, as some sort of dialogue between the utilities and the vendors, and these designs will therefore be in close correspondence with the utility requirements. On the other hand, it is quite obvious that the designs that have been, or are being, developed independently, will also be evaluated against these requirements, and modified to meet the functional requirements.

Hence, these functional, or top-tier, requirements will represent a common denominator of the design objectives for the different reactor concepts under development throughout the world, even though the design solutions may vary.

More details on the utility requirements documents being developed are presented in Chapter 2, together with a deliberation on some specific new safety requirements.

1.2 DESIGN OBJECTIVES FOR ADVANCED LWRs

1.2.1 Market situation

The market for nuclear power plants has been rather limited in recent years; very few new plants have been ordered during the last two decades. Eventually, this situation may change, since nuclear power does represent an important option for meeting a worldwide growth in energy demand in the years to come. At that point in time, the new plants will definitely not be replicates of present-day designs (of plants that are currently in operation), but will, beyond doubt, be of more advanced designs.

From a nuclear vendor, and perhaps also from a power utility, point of view, construction of new plants exactly as previous ones ("repeats") would yield a number of advantages; it implies reduced engineering and construction work, and significant cost savings. But, a plant design cannot live forever; it has to be "up graded" to implement new technology developments and new generations of equipment, to incorporate improvements based on experience gained from construction, commissioning and operation of the plants, and to adapt to regulatory requirements. There is also an ambition, by both vendors and utilities, to improve their product, making the nuclear plants more "forgiving", more reliable and easier to operate while keeping them economically competitive.

From the point of view of competitivity, striving for excellence is a must for the vendors, but it is also important for the utilities, from a public perception point of view; good averages are necessary but not sufficient, since the public will always base their judgement on the poorest performers. This was underlined by Lord Marshall, then chairman of WANO (World Association of Nuclear Operators), in a statement at its general meeting in Atlanta in April 1991: "it is no good having 99% of nuclear plants safe if the other 1% have low standards."

This implies that there has been, and will always be, a "continuous" development of new, more "advanced" designs. Vendors and utilities are envisageing a revival of a nuclear market within this decade, and the industry is preparing itself; a rather large number of "advanced" designs are under development. The majority of these new designs will introduce only small to moderate design modifications, but a few of them may introduce quite new approaches.

1.2.2 Improvement needs

The safety level of most nuclear power plants currently in operation is high per se and also in comparison with other energy sources. Safety remains an important issue for future nuclear power plants; plant simplification and reliability improvement are also aspects of importance for new, advanced designs. The safety of present-day modern plant designs is based on interactions between redundant and diverse safety systems, as a result of numerous system add-ons over the years to meet escalating safety requirements. From the technical point of view, this has represented a challenge to plant designers and operators with respect to ensuring reliable and safe operation. From the public perception point of view, a simpler design would make it easier to disseminate information on and explain plant performances. Besides, simplification tends to reduce number of components and building volumes; it helps control investment costs.

If safety systems are given a too complex design, this will obviously affect the economy of the power plant. Safety-grade systems and components require comprehensive periodic testing, inspections and maintenance, and a large number of such systems and components would represent a significant work load on the utility personnel. Much of this work has to be carried out in connection with refuelling outages and therefore tends to extend the plant shutdown; i.e., the plant economy is affected.

Escalating regulatory requirements in the past have to a large extent been a root cause to unwanted plant complexity, and lack of stability of regulatory requirements is in many countries considered as a problem. Development of appropriately updated and consistent regulatory requirements, including resolution of open licensing issues, and establishment of acceptable severe accident provisions, preferably on an international/global level, - given the international character of nuclear power needs and the worldwide potential effects of accidents, - would provide important support in the development of advanced reactors. The advanced reactor designers will

aim at achieving designs that will be consistent with such "supra-national" regulatory requirements; utilities also have expressed their preference for a standardization of designs.

The economy of nuclear power, as well as its safety, has been under much debate, - both in the public and within the industry, - and this explains why vendors and utilities strongly emphasize the need for simplification and improved reliability.

From the utility point of view, the new, advanced reactors should not depart too much from currently operating ones; preferably, the design modifications (or improvements) of present-day plant designs should be modest with a strong emphasis on maintaining "proven designs", thereby avoiding need for additional experimental verification and radical re-thinking of the operation management. If modifications and design changes were too large, with too much departure from, or supplements to, current designs in various aspects, part of the operating experience could be lost, and this must be factored in at the utility level when making the decision to order.

1.2.3 Desired characteristics of advanced designs

The design goals for advanced reactors are to a large extent focusing on a rather small set of typical common considerations, sometimes differentiated by supplementary "national" priorities.

The two most fundamental goals are related to plant simplification and to adequate operational margins - both with the prime objective of making plant operation simpler, from the point of view of the operator. Simplification improves operator comprehension of plant functions and implies that the actual operating conditions in the plant can be supervised more easily, i.e., it makes it possible for the operator to devote more attention to keeping the plant in operation and generating electricity. Increased design margins, - basically increased thermal margin for the fuel and enlarged coolant inventory to increase plant thermal inertia, - aim at providing more time for the operator to assess what actions need be taken in the event of disturbances or accidents; i.e., the operator in control is permitted to consider alternative actions carefully before acting (provision of "grace period"). Increased operational margins also result in a reduction of transients in the plant; fewer events and occurrences will lead to plant trips, and that, in turn, implies more reliable plant operation. Improvements in plant operability and reliability do bring significant benefits with respect to economy and also with respect to safety.

In the context of design margins, it may be noted that this involves reducing e.g. the core power density and fuel cladding surface heat rate to much lower levels than those prescribed by safety requirements in order to improve the "investment protection" and the "severe accident prevention" capability of the plant. Other aspects of the "thermal inertia" requirement can be the decision to increase water storage capability in plants relying on passive features, or providing diverse power supply such as combustion gas turbine in other plants to provide extended plant autonomy with respect to station blackout.

The safety objectives and strategies for advanced reactor designs will mostly correspond to those that were applied to present-day plant designs: - the general safety requirements will beyond doubt be the same (with a few supplements), and the defense-in-depth principle will be followed. Accident prevention by means of increased design margins and higher thermal inertia and lowered calculated probability for core damage is emphasized, and additional emphasis is given to accident mitigation by improved reactor containment strength and functional efficiency, - in order to ensure lower releases to the environment and small radiation doses beyond the plant grounds in accident situations.

In this context, it may be noted that the ALWR (advanced light water reactor) Utility Requirements Document that has been prepared by EPRI in the USA, introduces the term "good neighbour"; the plant shall be a good neighbour to its surrounding environment and population by minimizing radioactive and chemical releases - during normal plant operation as well as in accident situations. This corresponds closely to ambitions that have been expressed also in other countries, e.g., in Europe, - that possible releases from the plant shall be very low, even in accident situations involving significant core damage.

The rationale behind these ambitions is to ensure that, from a technical point of view, there would be practically no need for population relocation in the event of accidents. In other words, the aim is to provide a

technical basis for a significant simplification of the emergency planning (with no need for detailed evacuation planning and evacuation exercises) that constitutes the ultimate step of the defense-in-depth strategy.

Other features or design goals are of a more general nature. They relate to a design-for-construction approach with early completion of engineering and drawings - in order to improve and shorten the construction schedule over existing plants. Furthermore to use of proven technology, and standard equipment and components - in order to minimize investment risk, control costs, and take advantage of operating experience. They also encompass reviewing plant installations and maintenance procedures, etc. - to reduce operations and maintenance costs, to reduce occupational exposure and facilitate repair and replacement of equipment; and in-plant working environments, e.g., for the plant operators in the main control room, taking human engineering factors into consideration. The traditional requirements on utilization of high quality materials and application of adequate quality assurance programmes for the design and construction are of course maintained.

In short, the design objectives of advanced reactor designs aim at improvement of a variety of performance and other aspects:

- efficient treatment of licensing issues, and short, well defined engineering period, at the front end of the project;
- efficient construction work and confidence in set startup time during the construction period;
- high reliability, flexible operation, good maintainability, simplified operator work, low occupational exposure, good economy (achieving a clear advantage compared with alternatives) during subsequent commercial plant operation.

1.2.4 Other acceptability considerations

Looking back to the advanced reactor features mentioned above, the "good neighbour" policy needs to be substantiated further. Beyond showing the people living near the plant that normal or perturbed operation will have no negative impact on their everyday life, it is deemed necessary to explain that, even if the situation inside the containment were significantly degraded, consequences beyond the site boundary would be limited to below that requiring countermeasures for public protection.

All advanced designs under development address this consideration; designers are striving at minimizing the probability of core damages and the potential for degraded conditions inside the containment, by increased operational margins and by system improvements. To this end, probabilistic safety assessment (PSA) methodologies are applied extensively to check the safety and reliability of plant designs, as a supplement to analyses on deterministic basis. Such assessments represent extremely valuable engineering tools for comparing different design solutions, for analyses of plant functions and arrangements in order to identify design weaknesses, as well as for planning of test intervals and maintenance periods. The accuracy of a PSA depends very much on the modelling (representation) of the plant and its interacting systems, on the experience background and on component data; uncertainties in data and modelling will limit the accuracy at very low probability levels. Therefore, the results of PSAs are best used in a comparative way and should not be considered as absolute figures, in particular when dealing with very low probability figures they must be used with care.

So, designers are doing their utmost to prevent accident sequences from propagating to core damage and associated challenges to the containment system, considering also events and sequences beyond the current licensing design basis events (DBEs). Building upon the excellent resistance capabilities of current plants with respect to accident sequences leading to significant degradation of the core, and melting, and subsequent fission product release into the containment, severe accident challenges will be explicitly considered in the design of future plants. A practical consequence is a need for enhanced containment strength - either to the extent that it can withstand any conceivable load condition that may follow a core melt, or by adding equipment for filtered venting for overpressure protection. The Finnish General Design Criteria include a requirement for such measures; other European countries, such as Sweden, have adopted a more indirect approach by requiring installation of equipment for filtered venting of the reactor containment.

Based on inherent plant characteristics, or guidance from widely accepted documents such as those released by the IAEA, vendors have proposed design provisions or features to address relevant severe accident challenges in their design. The IAEA-TECDOC-682 "Objectives for the Development of Advanced Nuclear Plants" and IAEA-TECDOC-801 "Safety Principles for the Design of Future Nuclear Plants" (see Chapter 2),

recommendations issued by the French and German safety authorities, as well as severe accident policies defined by the US NRC and the Finnish Safety Authorities are examples of basic documents used by vendors when addressing severe accidents in their designs.

In some cases, probabilistic safety assessment (PSA) methodologies are used to identify challenges that have the potential for leading to unacceptable consequences and to address these challenges in a way which is consistent with otherwise defined acceptance criteria. When such an approach is adopted, it is stressed that PSAs are (or should be) used in a very comprehensive way. When dealing with very low probability values, great care has to be taken to make sure that quantification of improvements in plant response can be readily justified (e.g. by comparison with a reference evaluation made with the same methodology). Also, when screening challenges to be addressed or not, sensitivity studies showing that no cliff-edge effects exist are important to justify that a high confidence level supports the design decision.

Though PSAs, from an engineering standpoint, are powerful tools that are appropriate to the selection of severe accidents to be addressed in the design, as was discussed at the IAEA Technical Committee Meeting on Identification of Severe Accidents to be Addressed in the Design of Future Nuclear Power Plants in October 1995, other approaches can or do exist, relying on more deterministic requirements and recommendations. This is for instance the case for the common French-German Regulatory Requirements for Next Generation Plants, or equivalent Finnish Requirements. In such cases, designs are, or could be, requested to provide specific design features to address clearly identified challenges notwithstanding their significance in PSAs.

1.3 OUTLINE OF THE DEVELOPMENT OF ADVANCED LWRs

1.3.1 A general overview of advanced design developments

Over the past three decades, thousands of reactor-years of operating experience have been accumulated with the current economic and reliable nuclear energy systems. New generations of nuclear power plants have been, or are being developed, building upon this background of success and applying lessons learned from the experience of operating plants; hence, the new, advanced designs are anticipated to become even more safe, economic and reliable than their predecessors.

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

Advanced designs comprise two basic subcategories. The first is called evolutionary designs and encompasses direct descendants from predecessors (existing plant designs) that feature improvements and modifications based on feedback of experience and adoption of new technological achievements, and possibly also introduction of some innovative features, e.g., by incorporating passive safety features. Evolutionary designs are characterized by requiring at most engineering and confirmatory testing prior to commercial deployment. The other category consists of designs that deviate more significantly from existing designs, and that consequently need substantially more testing and verification, probably including also construction of a demonstration plant and/or prototype plant, prior to large-scale commercial deployment. Some of these designs introduce rather radical changes in design or configuration and these may be called innovative designs.

Advanced designs presently under development comprise three basic types:

- water cooled reactors, utilizing water as coolant and moderator;
- fast reactors, using liquid metal, e.g. sodium, as coolant; and
- gas-cooled reactors, using gas, e.g. helium, as coolant and graphite as moderator,

which are briefly reviewed in the following, in order to provide an overview of all the nuclear activities in the world.

Advanced water cooled reactors

Most - about 90% - of the nuclear power reactors that are currently in operation, are water cooled reactors, and most of the advanced designs are also water cooled reactors. The water reactors are of two basic types: LWRs with ordinary water as moderator, comprising 78% of the total number of nuclear units in operation, and HWRs which use heavy water, accounting for about 8%. The LWRs are in turn subdivided into boiling and pressurized water reactors (BWRs and PWRs). Advanced LWRs (ALWRs) are developed along two lines: large units of 1000-1500 MWe, and medium size units of about 600 MWe. The large units are generally building closely on predecessors, whereas the medium size units introduce new features, such as passive safety systems and plant simplifications, aimed at defeating the economic disadvantage of their smaller size (the economy of scale).

Some examples of large, evolutionary ALWRs are: the ABWR of General Electric (GE), USA, in co-operation with Hitachi and Toshiba, Japan; the APWR of Westinghouse (W), USA and Mitsubishi, Japan; the BWR 90 of ABB Atom, Sweden; the EPR of Nuclear Power International (NPI) - a joint company of Framatome, France and Siemens, Germany; the ESBWR of GE, USA; the SWR (or BWR) 1000 of Siemens; the System 80+ of ABB Combustion Engineering, USA; the WWER-1000 (V-392) of Atomenergoproject and Gidropress, Russia, and the KNGR of KEPCO, the Republic of Korea. Among the medium size ALWRs, four typical designs are: the AP-600 of W, USA; the AC-600 of China National Nuclear Corporation, China; the MS-600 of Mitsubishi, Japan; and the WWER-640 (V-407) of Atomenergoproject and Gidropress, Russia. Many of the large and medium size designs above are already available for commercial deployment, or will be within a number of years; the situation for innovative designs, such as: ISIS of Ansaldo, Italy; PIUS of ABB Atom, Sweden; SPWR of JAERI and IHI, Japan; and VPBER-600 of OKBM, Russia, is more unclear due to their need of experimental verifications and the associated need of more funds for the development.

HWR technology has also proven to be economic, safe and reliable. A mature infrastructure and regulatory base has been established in several countries, notably in Canada, the pioneer in the development of the HWR concept. Two types of commercial pressurized heavy water cooled reactors have been developed, the pressure tube and the pressure vessel versions, and both have been fully proven. HWRs with power ratings from a few hundred MWe up to approximately 900 MWe are available. The heavy water moderation yields a good neutron economy and has made it possible to utilize natural uranium as fuel which leads to lower fuel costs compared with LWRs. The amount of fissile material is quite limited, however, and the pressure tube and pressure vessel designs are therefore using on-load refuelling to achieve adequate reactivity for the plant operation. The effectiveness of this on-load refuelling has been successfully demonstrated; the annual and lifetime load factors of most of the pressure tube HWRs have been among the best of all commercial reactor types. Safety performance has also proven to be very good.

The continuing design and development programmes for HWRs in Canada are primarily aimed at reduction of plant costs and at an evolutionary enhancement of plant performance and safety. In Canada, three advanced versions; the 450 MWe CANDU-3 (recently put on hold), the 700 MWe CANDU-6 and the 900-1300 MWe CANDU-9, are under development within the evolutionary programme.

Also under development is an advanced 500 MWe HWR in India, and construction of such units is planned. This HWR design takes advantage of experience feedback from the 220 MWe HWR plants of indigenous design operating in India.

Fast reactors

Liquid metal cooled fast reactors (LMFRs) have been under development for many years in a number of countries, primarily as breeders, and the successful design, construction and operation of several plants, such as the 234 MWe PFR in the UK, the 233 MWe Phenix in France, the 70 MWe BN-350 in Kazakstan, the 560 MWe BN-600 in Russia, the 1200 MWe Superphenix in France and the 280 MWe Monju in Japan, has provided an extensive experience basis of more than 200 reactor years for further improvements. The fast reactors use fast neutrons for sustaining the fission process, and they actually produce fuel, as well as consuming it; plutonium breeding allows fast reactors to extract sixty times as much energy from uranium as thermal reactors do. Their capability of producing fissile material may become indispensable in the longer term if the deployment of nuclear power is increased substantially in the decades to come. Fast reactors may also contribute to burning of plutonium and to reduction of the required isolation time for high-level radioactive waste by burning of long-lived transuranic radioisotopes. The further development of fast reactors focuses on improving plant reliability and economy, on improvements of fuel burnup and fuel recycling technology to reduce the amounts of radioactive waste. Examples on development concepts are the BN-800M in Russia, the DFBR in Japan, the PFBR in India, the advanced European EFR, and the ALMR developed by GE in the USA.

Gas cooled reactors

Gas cooled reactors have been in operation for many years; in the United Kingdom (UK) the nuclear electricity is mostly generated in CO_2-cooled Magnox and advanced gas cooled reactors (AGRs). Development of high temperature reactors (HTGRs) with helium as coolant, and graphite as moderator, has also been going on for a long period of time. A number of prototype or demonstration plants have been built, without complete success, however. The inert He gas and a special fuel design enable it to operate at temperatures that are considerably above those in water cooled reactors; hence, it can supply steam for conventional steam turbine generators at much higher temperature (and pressure), or high-temperature process heat for special applications. Further HTGR development concentrates on improved plant performance and life extension studies, and in particular, much effort is devoted to the direct gas-turbine cycle which may yield very high thermal efficiency and low energy cost. A demonstration plant for testing of high temperature process heat applications is under construction in Japan, and construction has begun in China on a test reactor.

1.3.2 Overview of advanced light water reactor programmes

A main focus of development efforts in the most industrialized countries is on large size LWR units, with power outputs well above 1000 MWe; typically aiming at achieving certain improvements over existing designs. Alterations and modifications to a specific design are generally kept as small as possible taking maximum advantage of successful proven design features and components while taking into account feedback of experience from licensing, construction, commissioning and operation of the water cooled reactor plants currently in operation. The design improvements span a wide range but better economics, increased reliability, more user-friendly, and confirmed safety represent rather common denominators for the new designs. Smaller ALWRs are also being developed in a number of countries, in most cases with a great emphasis on utilization of passive safety systems and inherent safety characteristics. A survey of the advanced light water reactor designs presented and discussed in this report is presented in Table 1-I.

Important programmes in development of advanced light water reactors were initiated in the mid-1980s in the United States. In 1984, the Electric Power Research Institute (EPRI), a research organization of US utilities, in cooperation with the US Department of Energy (DOE) and with participation of US nuclear plant designers, initiated a programme to develop utility requirements for advanced light water reactors to guide ALWR design and development. Several foreign utilities have also participated in, and contributed funding to, the programme. Utility requirements were established for large BWRs and PWRs having power ratings of 1200 to 1300 MWe, and for mid-size BWRs and PWRs having power ratings of about 600 MWe.

Table 1-I Survey of advanced light water reactor designs under development

	Name	Type	Size [MWe]	Supplier/designer	Design status*
Evolutionary, large-size plants	APWR	PWR	1 300	Westinghouse, USA & Mitsubishi, Japan	Conceptual design
	ABWR	BWR	1 300	General Electric, USA****	Detailed design
	BWR 90	BWR	1 200	ABB Atom, Sweden	Detailed design
	EP 1000***	PWR	1 000	Westinghouse, USA, Genesi, Italy, EUR	Basic design
	EPR	PWR	1 500	Nuclear Power International (NPI)	Basic design
	ESBWR**	BWR	1 190	General Electric, USA	Preliminary design
	KNGR	PWR	1 350	Korea Electric Power Corp., Republic of Korea	Basic design
	Sizewell C	PWR	1 250	National Nuclear Corp. (NNC), UK	Conceptual design
	System 80+	PWR	1 350	ABB Combustion Engineering, USA	Detailed design
	SWR 1000	BWR	1 000	Siemens, Germany	Conceptual design
	VVER-1000 (V-392)	PWR	1 000	Atomenergoexport/Gidropress, Russia	Basic design
Evolutionary, medium-size plants	AC-600	PWR	600	China National Nuclear Corp. (CNNC), China, Peoples Rep.	Conceptual design
	AP-600	PWR	600	Westinghouse, USA	Detailed design
	HSBWR	BWR	600	Hitachi Ltd., Japan	Basic design
	MS-600	PWR	600	Mitsubishi, Japan	Detailed design
	VVER-640 (V-407)	PWR	640	Atomenergoexport/Gidropress, Russia	Detailed design
Design concepts requiring substantial development	ISIS	PWR	300	Ansaldo Spa., Italy	Conceptual design
	JPSR	PWR	630	Japan Atomic Energy Research Institute (JAERI), Japan	Design study
	PIUS	PWR	650	ABB Atom, Sweden	Basic design
	SPWR	PWR	600	Japan Atomic Energy Research Institute (JAERI), Japan	Conceptual design
	VPBER-600	PWR	630	OKBM, Russia	Conceptual design

* The design status classification refers to new IAEA-TECDOC on Terms for Describing New, Advanced NPPs
** ESBWR (and JSBWR in Japan) is enlarged version of GE's SBWR design
*** EP 1000 (in Europe & SPWR in Japan [M]) represents enlarged versions of AP-600
**** This refers to the U.S. version; the ABWR was developed jointly by GE and Hitachi/Toshiba, Japan, and two Japanese ABWR plants are already in operation.

In 1986, the US Department of Energy, in cooperation with EPRI and reactor design organizations, initiated a design certification programme for evolutionary plants based on a new licensing process, followed in 1990 with a design certification programme for mid-size plants with passive safety systems. The new licensing process allows nuclear plant designers to submit their designs to the US Nuclear Regulatory Commission (US NRC) for design certification. Once a design is certified, the standardized units will be commercially offered, and a utility can order a plant confident that generic design and safety issues have been resolved. The licensing process will allow the power company to request a combined license to build and operate a new plant, and as long as the plant is built to pre-approved specifications, the company can start up the plant when construction is complete, assuming no new safety issues have emerged.

Four advanced reactor designs developed in the US have been submitted to the US NRC for certification under the US DOE ALWR Programme. Two large evolutionary plants, the System 80+ of ABB Combustion Engineering and the ABWR of General Electric, received Final Design Approval in 1994 and are expected to receive Design Certification in 1997[1] .The Westinghouse 600 MWe AP-600 is under review by the NRC, and a Final Design Approval is expected by March 1998. Up to mid-1996, the 600 MWe simplified BWR (SBWR) developed by General Electric was also under review, but then General Electric stopped work on the 600 MWe version and shifted its emphasis to a unit with larger output. The development of both these mid-size designs has been governed by a key guideline: new features should need no more than engineering and confirmatory testing before commercial deployment; they must not lead to a requirement of building and operating a prototype or demonstration plant. A first-of-a-kind engineering (FOAKE) (the detailed design needed to verify cost and schedule) programme, which was authorized by the Energy Policy Act of 1992, has been completed for the ABWR in 1996, and similar work on the AP-600 is scheduled for completion in 1998.

In Europe, some evolutionary plant designs are currently under construction or in operation, while others are in varying phases of regulatory review. The British Sizewell-B, a 1250 MWe PWR, which is an evolutionary design, based on the WPSS design of Westinghouse, and the N4 model in France, a 1450 MWe PWR, which derives directly from the standardized 1300 MWe P4 series, have been taken into operation in 1995 and 1996, respectively.

Framatome and Siemens have established a joint company, Nuclear Power International, which is developing the EPR, a new advanced PWR of 1500 MWe (gross), that will be reviewed jointly by the French and German safety authorities. This is anticipated to result in a practical harmonization of the safety requirements of two major countries, which could later be enlarged to a broader basis.

Siemens is also, together with German utilities, engaged in the development of an advanced BWR design, the SWR-1000, which will incorporate a number of passive safety features, for initiation of safety functions, for residual heat removal and for containment heat removal.

In Sweden, ABB Atom, with involvement of the utility Teollisuuden Voima Oy (TVO) of Finland, is developing the BWR 90 as an upgraded version of the BWRs operating in both countries. The company has also for quite some time been pursuing the development of PIUS, an innovative, super-safe "pool-type" PWR, for which the development activities are at a hold for the time being.

An adaptation of the AP-600 and SBWR designs to meet the requirements of the European utilities (EURs) and the need for plants with higher power outputs is underway in Europe. The two programmes are named EPP and ESBWR, respectively.

In the Russian Federation, design work on the evolutionary V-392, an upgraded version of the WWER-1000 (a 1000 MWe PWR), has been started, and another design version is being developed in cooperation with Imatran Voima Oy (IVO) of Finland. The Russian Federation is also developing an evolutionary WWER-640 (V-407) design which incorporates passive safety features, as well as a more innovative, integral design, the VPBER-600. Construction of pilot units of the WWER-640 and the upgraded WWER-1000 are planned.

In the Republic of Korea, an effort is underway since 1992 to develop an advanced design termed the Korean Next Generation Reactor (KNGR), an evolutionary large 4000 MWth PWR design. The basic design is

[1] Subsequently the System 80+ design and the ABWR design received Design Certification in May 1997.

In the Republic of Korea, an effort is underway since 1992 to develop an advanced design termed the Korean Next Generation Reactor (KNGR), an evolutionary large 4000 MWth PWR design. The basic design is currently being developed by Korea Electric Power Corporation (KEPCO) and the Korean nuclear industry. The goal is to complete a detailed standard design by the year 2000.

In Japan, the Ministry of Trade and Industry (MITI) is conducting an "LWR Technology Sophistication" programme focusing on development of future LWRs and including requirements and design objectives. Development programmes for a Japanese simplified BWR (JSBWR) and PWR (JSPWR) are in progress jointly involving vendors and utilities, and the Japan Atomic Energy Research Institute (JAERI) has been investigating two advanced design concepts with emphasis on passive safety systems, the JAERI Passive Safety Reactor (JPSR) and the System-Integrated PWR (SPWR).

Other examples of development activities in Japan are two large, evolutionary designs: the Westinghouse-Mitsubishi advanced pressurized water reactor APWR (a 1350 MWe PWR), and the General Electric-Hitachi-Toshiba Advanced Boiling Water Reactor ABWR (a 1350 MWe BWR). In this context, it may be noted that, one ABWR unit - the Kashiwazaki Kariwa unit 6 - was taken into commercial operation in November 1996, and another - Kashiwazaki Kariwa unit 7 - was connected to the grid in December 1996[2] . JAPCO is planning construction of two APWR units at the Tsuruga site.

In China, the AC-600 advanced PWR which incorporates passive safety systems for heat removal, is under development by the Nuclear Power Institute of China.

All of these ALWRs incorporate significant design simplifications, increased design margins, and various technical and operational procedure improvements, including better fuel performance and higher burnup, a better man-machine interface using computers and improved information displays, greater plant standardization, improved constructability and maintainability.

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[2] Kashiwazaki Kariwa unit 7 subsequently went into commercial operation in July 1997.

CHAPTER 2. UTILITY REQUIREMENTS AND SAFETY OBJECTIVES FOR ADVANCED WATER COOLED REACTORS

2.1 DEVELOPMENT OF UTILITY REQUIREMENTS DOCUMENTS

In most countries of the world, the owners/operators of nuclear power plants, basically utilities or groups of utilities, have in the past relied on the advices and experience of the suppliers for deliveries of turn key nuclear power plants or nuclear steam supply systems (NSSS) for such plants. The plant designs were launched by the suppliers and/or architect engineers based on their experience, and the utilities mainly limited their involvement in the design and construction to certain review work and to training on the job. The role of the operating utility as the responsible entity for plant safety vis-a-vis the regulatory authorities made it necessary for utilities to build up their own engineering capability to enable performing design engineering beyond the operation and maintenance activities. The licensing of nuclear power plants normally followed internationally established rules taking into account also national codes and standards of the country intending to build the plant. When specific national codes did not exist, however, the codes and standards of the country of origin were frequently taken as the basis.

Over the past three decades, utilities have accumulated a lot of experience from their operating nuclear power plants, some good and other things less good. New or modified requirements that the plant has to be adapted to, have very often been imposed during the life time of the plant; the adaptation has often required certain re-engineering as well as new design approaches. The utilities started to specify their special desires for new nuclear plants, reflecting their specific experience, in bid invitation specifications; this can be seen as the first steps to the development of utility requirements for future advanced reactor concepts. The utilities realized that it would be beneficial to bring together all the experience gained, and established so called owners' groups (e.g., BWR and PWR owners' groups) in order to exchange operating experience with a specific type of reactor.

In the past, the degree of standardization of nuclear power plant design attained has generally been limited, in particular so in the United States. In France, a much higher degree of standardization was achieved by ordering and building series of nuclear power plants of the same design, yielding shorter construction times and lower costs. On the basis of such facts, the ministry of international trade and industry (MITI) in Japan initiated a long term "LWR Standardization Program for Improvements" in the mid 1970s, and in the United States, a major utility project was launched in the mid 1980s to try establishing a complete set of utility requirements for a next generation of standardized nuclear power plants.

2.1.1 EPRI Utility Requirements Document (URD)

In the US project, the utilities are leading an industry wide effort to develop the technical bases for the design of advanced light water reactors (ALWRs). The effort is being managed for the US electric industry by the Electric Power Research Institute (EPRI), in close co-operation with the US Department of Energy (DOE), and it includes participation and sponsorship of several international utility companies.

The purpose of the requirements document being developed in this multi-national effort, is to present a clear, complete statement of utility desires for design, construction and performance of a next generation of nuclear power plants. The main objectives for the establishment of the requirements document were to:

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

The culmination of this effort is the EPRI advanced light water reactor (ALWR) Utility Requirements Document. This document is the cornerstone of the US ALWR program which covers development of several LWR designs for the future.

The EPRI ALWR utility requirements document (URD) addresses the entire plant up to the grid interface, including nuclear steam supply system and balance of plant, and it applies to both boiling water reactors (BWRs) and pressurized water reactors (PWRs). The requirements intend to provide improved and standardized versions of ALWRs, which eliminate most of the problems and inefficiencies associated with some of the existing designs; assure a simpler, more "forgiving" plant design which is excellent in all respects, including safety, performance, constructibility, and economics. The first complete version of the URD was issued in 1990 and contains more than 14 000 detailed requirements for ALWR designs. Since then, there have been seven revisions to the URD.

The EPRI URD is organized in three volumes. Volume I summarizes ALWR programme policy statements and top tier requirements. Volumes II and III present the complete set of top tier and detailed requirements for specific ALWR design concepts. Volume II covers "evolutionary" ALWRs. These are simpler, much improved versions of existing LWRs, with a power output up to 1350 MWe, employing conventional, but significantly improved, active safety systems. Volume III covers "passive" ALWRs, greatly simplified, smaller (i.e., of reference size 600 MWe) plants which employ primarily passive means (i.e. natural circulation, gravity drain, and stored energy) for essential safety functions. Two passive reactor designs are addressed in Volume III, the passive BWR with pressure suppression containment and the loop type passive PWR with dry containment. While these Volume III designs are in the final stages of completion, they do utilize existing LWR experience and evolutionary ALWR requirements, and are expected to offer substantial advantages in constructability and operability as well as the potential to surpass the high ALWR safety standards.

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

Since its publication in 1990, the URD has been used in several design projects. The programmes have been implemented to assure conformance between the requirements and the designs. The ALWR design projects have shown that the use of utility requirements such as those incorporated in the URD represents an essential element within the process of developing new LWR designs.

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

2.1.2 European Utility Requirements Document (EUR)

The European Union is intended to be an open, common market, and the market barriers are lifted progressively, as the European Union market rules are enforced, allowing wider competition. With respect to nuclear energy, however, Europe is today still partitioned into discrete countries, since the nuclear safety rules and regulations are promulgated by independent, national regulatory bodies. This partitioning is re-inforced by the fact that nuclear energy in many countries is considered as being of "strategic" value, making it important to maintain a "domestic" nuclear capability.

Considering this perspective, a group of major European utilities, from Belgium, France, Germany, Italy, Spain, the Netherlands and the United Kingdom, (in 1996, expanded with Finland and Sweden), decided to take a lead role in defining the main features of future plants, suitable for all parties, and they initiated at the beginning of 1992 the development of a common requirements document, the European Utility Requirements (EUR). At first, they agreed to propose a common set of safety requirements to be discussed and eventually endorsed by the regulators, in an effort to achieve harmonization of safety requirements between the different countries. Then the EUR scope was extended so that it could allow the development of competitive, standardized designs that would match the conditions in Europe and be licensable in the respective countries. In this context, it should be noted that the utility requirements are not intended to offset the national hierarchies of laws/acts (e.g., national Atomic Energy Acts, stipulations/ordinances, rules and guidelines); the plant design will always have to duly consider and comply with the national laws/acts. Nevertheless, the national laws could possibly be reviewed to incorporate pertinent utility requirements.

The EUR is being employed for the design of the european pressurized water reactor (EPR) which is under development as a joint effort between French and German utilities and the vendors Framatome and Siemens, through a joint company nuclear power international (NPI). The requirements are also employed for the design of the european passive PWR (EPP) and the european simplified BWR (ESBWR) development projects in which the group of European utilities is taking part. The level of detail specified in the EUR is generally less than in the EPRI URD, giving some flexibility to the plant designer with respect to specific design solutions and special features. The first issue of the EUR was released in 1994, and submitted to a thorough review by peer utilities and vendors. A revised version was released at the end of 1995.

Since the EPRI URD, which is a really comprehensive set of utility requirements, had already been established in the United States, great care is taken to avoid introducing differences that could not be justified either by specific objectives of the European utilities or specific European factors such as regulatory rules or environmental conditions.

The EUR document will be the reference user's document for LWR plants to be built in Western Europe beyond the turn of the century. It will be stabilized when the discussions with the different safety authorities have been completed.

The EUR document is structured in four volumes containing the various levels of requirements. The main policies and top tier requirements as major design objectives are incorporated in Volume 1, and Volume 2 delineates all generic nuclear island requirements which are not related to any specific design. Volume 3 contains specific nuclear island requirements related to a specific design which is of particular interest for some of the participating countries (each such design will have a separate Volume 3), and Volume 4, finally, will specify generic conventional island requirements for which the outline is not yet finally determined. The involvement of the safety authorities in the discussions, to have their desires considered in the EUR documents from the very beginning, is seen as a matter of particular importance for the future applicability of the requirements.

2.1.3 Japanese Utility Requirements Document (JURD)

As noted above, a Japanese LWR standardization program for Improvement was initiated in the mid 1970s as a joint effort between the Government and the industry, under the leadership of the Ministry of International Trade and Industry (MITI), with the objective of improving and standardizing the LWR designs to enhance the reliability and plant availability and to reduce occupational exposures. The programme included three phases; the first and second phases which were carried out during 1976 to 1980, were mainly addressing possible improvements of the operating plants, whereas the third phase which started in 1981 aimed at establishing a Japanese light water reactor design based on own technology, including the reactor core, as well as systems and components.

Based on these programmes, two advanced light water reactor designs, the advanced boiling water reactor (ABWR) and the advanced pressurized water reactor (APWR) have been developed. Two ABWR units have also been built at Kashiwazaki-Kariwa, as units No. 6 &7; the first of these was connected to the grid in January 1996, and the second in December 1996. A firm decision on construction of the APWR has not yet been made, but the Japan Atomic Power Co (JAPCO) has announced plans on building a twin unit at the Tsuruga site.

Studies and investigations for LWR designs to follow the ABWR and APWR are being carried out by MITI and by nuclear industry groups.

The design philosophy of future LWRs, for deployment between 2010 and 2030, will be formulated by MITI, considering the social environment of Japan, enhancing economic efficiency while also pursuing enhancement of safety and regulation related to adopting the newest technologies, and taking into consideration international trends.

An ABWR evolutionary programme was initiated in 1990 by the Japanese BWR utilities, jointly with Hitachi, Toshiba and GE on possible evolutionary improvements of the ABWR taking advantage of new technology. The target for electrical output is 1500 MW, but the possibility for an increase to 1700 MW is also being evaluated.

The PWR utilities, together with Mitsubishi and Westinghouse, initiated in 1994 a programme to develop the next generation PWR, the successor to the APWR. In this development, comparative studies of active and passive safety systems, as well as combinations of active and passive systems are being performed. Design requirements and conceptual design as well as research and development of applicable component technologies are being studied in the programme.

2.1.4 Korean Utility Requirements Document (KURD)

The purpose of the Korean Standard Requirements Document (KSRD) was to define the requirements for the next series of PWRs to be built in the Republic of Korea in the mid-to-late 1990s, the Korean Standard Nuclear Power Plant Design. This document covers the conventional PWR, with add on improvements appropriate for application in Korea. The KSRD is similar in scope to Volume II of the EPRI ALWR URD for PWRs, but contains some significant differences, which reflect the desires of the Koreans to establish their own design and construction capability. This document was completed in 1990.

The Korean efforts to develop user requirements for future plant designs began in 1993, as part of the next generation reactor development project. The objective of the development of these user requirements was to delineate necessary features and characteristics of future reactors suitable to the Korean conditions, so that the direction of future reactor development in the Republic of Korea can be determined.

The Top Tier requirements were developed and the reactor type was determined during the two year period from 1993 to 1994; the work is currently focusing on more detailed requirements and specific features.

The development of the user requirements is being carried out in parallel with the basic design of the Korean Next Generation Reactor (KNGR) in such a way that the requirements are established ahead of design work. Hence, the requirements are available prior to system design and reviews of the requirements regarding their applicability and usefulness are fed back to the developers of the user requirements.

In the current phase of the KNGR programme, user requirements for the design of major systems are being developed, with completion scheduled for 1998. In the next phase of the KNGR programme which will develop a detailed design, the user requirements will be reviewed and modified as the design of KNGR becomes more detailed.

2.1.5 Taiwan Power Company Requirements Document (TPCRD)

The Taiwan power company requirements document (TPCRD) was developed for the fourth Nuclear Power Plant Project, the Lungmen project. This document was an essential and integral part of the Bid specifications for the nuclear island, the turbine generator and the balance of plant of the plant to be offered. The requirements document covers evolutionary designs of BWR and PWR type and is in scope and detailed content very similar to the EPRI ALWR URD.

2.2 TOP LEVEL UTILITY REQUIREMENTS

2.2.1 General overview

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

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

The top level requirements of Volume 1 are largely of a general nature, dealing with generic issues that are valid for all types of ALWRs. The EPRI URD, however, includes also some requirements for specific reactor types, namely for the passive reactors that are part of the US ALWR development programme. In the EUR, requirements for specific reactor designs (of BWR and PWR types) are not yet included; they will be developed at a later stage.

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

2.2.2 EPRI ALWR URD requirements

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

Chapter 2 is entitled "ALWR program policy statements" and it delineates the policy on a number of nuclear power plant issues that to some extent are not directly related to design or safety requirements:

- emphasize simplification;
- provide significant design margin;
- take human factors into consideration;
- ALWR safety - protect both the public and the investment;
- include safety margin beyond ALWR design basis;
- regulatory stabilization - direct co-operation with the US NRC;
- plant standardization - develop technical specifications in sufficient detail with industry/NRC consensus;
- proven technology - successful technology shall be employed throughout the plant;
- maintainability - design from outset to make plant readily maintainable over its life;
- constructibility - aim is substantially improved construction schedule;

- Quality Assurance - effective performance oriented QA program to ensure high quality;
- ALWR economics - low busbar costs with sufficient cost advantage;
- ALWR sabotage protection - considered in the design; and
- ALWR good neighbour - plant shall be a good neighbour to its surrounding environment and population.

The top level ALWR design requirements with respect to design aspects are described in Chapter 3. They are broken down by function into four sub-chapters: safety design requirements, performance design requirements, constructability requirements and design process requirements. The content of these sub-chapters are discussed further below.

The economic goals for the ALWRs are outlined in Chapter 4, and Chapter 5 defines the role of the ALWR requirements with regard to implementation of the ALWR and plausible scenarios for this implementation.

From a designer point of view, the top level design requirements of chapter 3 include a lot of information, including specific design figures, which should be understood as mandatory and have to be complied with, independent of reactor type. Some examples drawn from the different sub-chapters may illustrate the quality of that specification.

Safety design requirements

1. Accident resistance

- Design shall include ample margins, e.g.:
  - Fuel design margin of 15% over and above regulatory fuel design requirements
  - PWR maximum vessel exit temperature of 316°C (600°F) [not really a requirement focused on safety]
- The safe shutdown earthquake (SSE) shall be 0.3g
- The operator shall have adequate time (30 minutes or more) to act after accidents/incidents

2. Core damage prevention

- With respect to core damage prevention the supplier has to demonstrate that the cumulative core damage frequency is less than 10^{-5} events per reactor year, including internal and external events.

3. Mitigation

- The licensing design basis shall provide H_2 control, and the H_2 concentration in the containment should not exceed 10% under dry conditions considering an amount of H_2 equivalent to that generated by oxidation of 100% of the active fuel cladding.
- Severe accident risk shall be evaluated in a PRA and the whole body dose at site boundary (approximately 0.5 miles (or 800 m) from the reactor) is less than 25 rem for release from severe accidents, the cumulative frequency of which exceeds $1x10^{-6}$ per reactor year.
- Containment integrity shall be warranted for all accidents considered in the design.

Performance design requirements

1. Plant characteristics

- The plant shall be designed to operate for 60 years.
- The plant should be capable of operating on a fuel cycle with a refuelling interval of 24 months.
- BWR peak bundle average burnup should reach 50 000 MWd/t U, for PWR the value should be 60 000 MWd/t U.
- The premature failure rate due to manufacturing defects shall be less than one in 50 000 fuel rods.

- Wet storage capacity for spent fuel resulting from 10 years of operation plus one core shall be provided.

2. Manoeuvring and transient response requirements

- The plant shall be designed for a 24 hour cycle with the following cycle profile: starting at 100% power, power ramps down to 50% in 2 hours, power remains at 50% level for 2-10 hours, and then up to 100% in 2 hours. The plant design shall permit this cyclic load for 90% of the days of each fuel cycle during the plant life.

3. Reliability and availability requirements

The following top-tier reliability and availability requirements apply:

- The plant shall be designed for an annual average availability of greater than 87 percent over the life of the plant.
- The plant shall be designed to achieve the following outage durations:
  - Planned outages: less than 25 days/year
  - Forced outages: less than 5 days/year
  - Major outages: less than 180 days/10 years
- The plant shall be designed so that a refuelling outage free from major problems can be conducted in 17 days or less (breaker to breaker) assuming 24-hour productive days.
- The plant shall be designed to limit the number of unplanned automatic trips to be less than one per year. In response to this requirement, the plant shall utilize a minimum number of plant variables for reactor trip signals consistent with plant safety and shall provide increased margin between the normal operating range and the trip set point of safety variables so that the number of plant trips resulting from normal operation activities is minimized.

2.2.3 EUR requirements

Based on the revision B of the EUR requirements some aspects are selected in a similar way as for the URD to show the coherence or the deviations of the two requirement documents to each other.

Safety design requirements

1. Accident resistance
   - **Earthquakes:** The design basis earthquake (SSE) is defined by a standard design acceleration and a set of seismic spectra reflecting the European conditions. A set of three European SSE spectra is given for hard, medium and soft soil. A standard design acceleration level of 0.25g is required for these three spectra.
   - **External explosion:** One standard design case (a triangular wave peaking at 100 mbar and lasting for 300 ms) is defined for an external explosion.
2. Core damage prevention
   - Core damage cumulative frequency shall be less than 10^{-5} per year and a considering both operation and shutdown states, and including internal and external events.
3. Mitigation
   - Severe accident risks shall be evaluated in a PRA and the cumulative frequency of exceeding the limiting release set for severe accidents with core degradation, shall be less than 10^{-6} per year.
   - The licensing design basis shall provide H_2 control, and the H_2 concentration in the containment should not exceed 10% under dry conditions considering an amount of H_2 equivalent to that generated by oxidation of 100% of the active fuel cladding.

- The containment system shall comprise a primary containment designed to withstand the mechanical and thermal loads arising from internal accidents, including severe accidents involving core damage, and to confine the radioactivity released during accidents as a leaktight structure (including also the penetrations), and a secondary containment designed to collect part or all of any releases from the primary containment.
- With respect to the potential consequences of hydrogen accumulation in the primary containment following a severe accident, it is required that the effects of a global adiabatic deflagration of 10% dry concentration of H_2 shall be considered.
- The design shall include measures to collect (corium spread area) and cool (through in-containment water storage) molten core materail ejected from the RPV.
- High pressure melt ejection events shall be eliminated by RCS depressurization system, and containment shall include measures to decrease pressure to 50% of peak value in 24 hours after an accident.

Performance design requirements

1. Plant characteristics
   - The design life of plant structures and non-replacable components such as the RPV shall be 60 years;
   - The core and NSSS of the standard plant shall be capable of producing its rated power under the combination of:
   - cores with 100% UO_2 and 50% MOX fuel assemblies mixed with UO_2 fuel for a refuelling interval between 12 and 24 months and maximum peak assembly burn up of 60 000 MWd/t of HM for UO_2 fuel and 45 000 MWd/t of HM for MOX fuel.
2. Availability targets
   - The plant shall be designed to achieve a high availability during its operational lifetime.
   - The main requirements are:
     - high overall availability of the plant (capability factor greater than 87%),
     - short planned outage duration (average refuelling and maintenance outage shorter than 25 days per year, refuelling only outage possible in less than 17 days, major plant outage shorter than 180 days per 10 years),
     - low level of unplanned outages (unplanned automatic scrams less than 1 per 7000 hours critical, unplanned capability loss factor less than 1.4%).
3. Release targets
   - Release rates for normal operation and incidents, "utility limits and targets" are judged to be appropriate to take into account national and international requirements and should be aimed for as part of implementing the ALARA concept. Release targets for severe accidents are referred to as "Limiting Release" in the EUR document. The limiting release value is intended to be so low that the societal consequences resulting from public health effects and contamination of soil and water will be limited; specifically, the EUR anticipate that realization of this target will imply:
   - minimal emergency protection action beyond 800 m from the reactor during early releases from the containment,
   - no delayed action (temporary transfer of people) at any time beyond about 3 km from the reactor,

Table 2-I Radioactive releases for one 1500 MWe LWR unit

	Utility limit	Target
Liquid discharges	GBq/a	GBq/a
Gaseous discharges: • Noble gases • Halogens and aerosols	 TBq/a GBq/a	 TBq/a GBq/a

- no long term actions, involving permanent (longer than 1 year) resettlement of the public, at any distance beyond 800 m from the reactor.
- In addition, it is stipulated that restriction on the consumption of foodstuff and crops shall be limited in terms of time scale and ground area. Further details are given in Volume 2 of the EUR document.
- Table 2-I above gives the main limits and targets related to liquid and gaseous releases in design basis category 1 and 2 conditions (normal operation and incidents), including maintenance and overhaul operations.

4. Targets for low activity solid radwaste
   - The total volume of the final solid radwaste produced by one plant should be less than $50m^3$ per 1000 MWe per year of normal operation.

2.2.4 Japanese requirements

The Japanese requirements have not yet been published in a systematic way, but some information can be found in conference papers, e.g., in the paper on "The Japanese Utilities' Requirements for a Next Century BWR" by M. Ishikawa, T. Anegawa, A. Komori and A. Omoto, Tokyo Electric Power Company (TEPCO), presented at ICONE-3 in Kyoto, Japan, in May 1995.

The utilities have formulated four fundamental desirable characteristics for the new plant designs; they shall be:

- economical - competitive with alternate power generation sources in terms of capital cost;
- benign to human - more human-friendly in terms of operation and maintenance;
- simple - simplicity in design will be important with respect to human resources; and
- flexible - to uncertainties in energy issues, in fuel cycle, trends in international safety standards, and other changes in infrastructure including human resources.

Commercial operation is assumed to begin in the 2010s, based on the projected power demand in Japan and the necessary development period. The power generation capacity will be about 1500 MWe (perhaps up to 1700 MWe) since this rating appears achievable by extrapolation of current technology.

The required top level plant characteristics are subdivided into three categories:

- Operability;
- Safety; and
- Core and fuel.

Operability (or performance design requirements)

1. Plant characteristics
   - The plant shall be designed to operate for 60 years.

- The plant design shall include measures to reduce maintenance activities in harsh environments and to reduce occupational exposure to less than that of the ABWR.
- The design shall consider the "N+2" design concept to reduce peak loads during annual inspections.

2. Manoeuvring and transient response requirements
   - Power level changes of ± 5 % shall be accommodated within 30 seconds.
   - The plant shall be capable of participating in grid frequency control with a maximum power variation range of ± 1.5 %, with a response time of 10 seconds or less.
   - The plant shall be designed for a 24 hour cycle with the following cycle profile: starting at 100% power, power ramps down to 50% in 2 hours, power remains at 50% level for 2-10 hours, and then up to 100% in 2 hours.
   - Plant startup time from hot standby to 90 % power shall not exceed 5 hours.

3. Reliability and availability requirements
   - The plant shall be designed so that a refuelling and inspection outage can be conducted in less than 40 days or less.
   - The plant shall be designed to prevent inadvertent reactor scram and engineered safety system actuation due to single failure of normal duty systems or single operator error.

Safety (safety design requirements)

1. Accident resistance
   - Design shall include ample margins,
   - The operator shall have adequate time (one hour in the case of transients, one day in the case of accidents) to act after accidents/incidents

2. Core damage prevention
   - The design shall use active safety systems for the short-term cooling and a passive system as a backup for active systems for long-term cooling under accident conditions.
   - The mean core damage frequency (CDF) for the design shall be equal to or lower than 10^{-5} events per reactor year, including internal and external events.

3. Mitigation
   - The conditional containment vessel failure probability (CCFP) shall be equal to or less than 0.1, by adopting accident management.
   - The design shall adopt severe accident countermeasures, both with respect to core damage prevention and containment vessel damage prevention measures, in particular to prevent damages in the direct containment heating (DCH) and ATWS sequences.

Core and fuel (mostly performance requirements)

1. Flexible handling of the fuel cycle
   - Design shall allow improvement of uranium utilization using uranium-saving technology approaches.
   - It shall be possible to increase burnup substantially to provide an average discharge burnup of about 70 GW d/t (from the current 45 GW d/t) - of particular interest for long-term cycle operation and once-through cycle (without reprocessing);
   - The design shall allow for easy handling of large plutonium inventories, i.e., the plant shall be capable of operating with a large amount (up to 100%) of MOX fuel;
   - The design shall allow variations in operating cycle length.

- The core design shall be simple, and reduce both core designer and operator work loads, without complicated operation management.
- The design shall ensure speedy refuelling.
- The design shall allow reduction in maintenance and inspection loads, without requiring use of complicated reactor internals or control rods.

2.2.5 Commonalties and differences

The utility requirements are intended to be, and should be used as, a basis for reduction of uncertainties in planning, design and operation of future nuclear power plants in the geographical region in which they will be applied. A comparison of the limited number of parameters extracted from Chapter 3 of Volume 1 of the EPRI URD with those from the EUR, as well as the Japanese requirements, shows that there is a clear coherence in many of the design requirements[1]. Among others, they all ask for:

- a design life time of 60 years,
- reliable and flexible operation, with high overall plant availability, low level of unplanned outages, short (or shortened) refuelling outages, good controllability (e.g., 100-50-100% load following capability), and extended operating cycles up to 24 months,
- increased margins to reduce sensitivity to disturbances (and number of safety challenges),
- improved automation and man-machine interface which together with the increased margins provides more time for operator to act in accident/incident situations, and reduces probability of operator errors,
- core damage frequency less than 10^{-5} per year and cumulative frequency of large releases following a core damage less than 10^{-6} per year, and
- design measures (e.g, improved RCS depressurization system, and in-containment water storage) to cope with severe accidents are requested.

With respect to subjects dealing with the licensing procedure and/(or) the impact of the new reactor to the environment, the requirements in Europe appear to be different from those in the United States, even though the deviation in some cases may arise from different ways of formulating similar goals. In one specific area, however, there is a distinct difference in requirements; the EUR include release targets that go beyond those given in the EPRI URD, in particular for severe accident situations. The defined goals with the "Limiting release" requirement of the EUR are:

- to limit emergency protection actions beyond 800 m from the reactor during early releases from the containment to a minimum,
- to avoid delayed actions (temporary transfer of people) at any time beyond about 3 km from the reactor,
- to avoid long term actions, involving permanent (longer than 1 year) resettlement of the public, at any distance beyond 800 m from the reactor, and
- to ensure that restrictions on the consumption of foodstuff and crops will be limited in terms of time and ground area.

The reason for these stipulations is the higher population density in Europe which makes it difficult and costly to implement adequate emergency procedures that include relocation of people, and the greater attention put on land use limitations. In order to get a new site with an advanced LWR licensed, the utility/utilities would have to convince the public of no need for precautions in case of any accident.

[1] In order to attain a better understanding of commonalties and differences between the various utility requirements documents the IAEA initiated in late 1996 a "Review of User Requirements for Future NPPs"

2.3 SAFETY OBJECTIVES

2.3.1 Current licensing and design basis requirements

The safety philosophy for the current generation of nuclear power plants is in most cases based on design measures to ensure that a certain scope of initiating events, defined as design basis events (DBEs), will not result in temperatures and pressures exceeding the acceptable levels stipulated by the authorities, when analysed in accordance with established rules and methodologies. The most challenging accident sequences are defined as design basis accidents. For water cooled reactors, failures of the reactor coolant pressure boundary and subsequent loss of coolant, the loss-of-coolant-accidents (LOCAs), represent accident sequences of particular interest. A large LOCA results in a significant pressure transient and a rapid loss of coolant, but even in the event of rupture of the largest pipe connected to the reactor pressure vessel (the maximum LOCA) this must not affect the integrity of the core and lead to loss of its coolability, resulting in fuel temperatures in excess of levels stipulated by the authorities. In order to limit the consequences, and prevent core heatup and degradation, safety systems with high redundancy and diversity are installed.

In the event of such an accident, however, the radiological impact to plant personnel and to the public living in the vicinity of the plant by the reactor coolant outflow from the primary circuit is limited by means of internal emergency measures and of the containment and its associated support systems. As an ultimate protection of the population, emergency plans involving measures such as sheltering and evacuation, have been developed.

In summary, the safety strategy for currently operating reactors is characterized by three levels of design measures, based on current licensing and design basis requirements:

1. Accident resistance is ensured by design margins, redundancy and diversity of safety systems, and in-service inspection and testing to assure reactor coolant pressure boundary (RCPB) integrity.
2. Core damage prevention is ensured by dedicated safety systems to meet regulatory requirements (regulatory fuel temperature limits shall not be exceeded in accident sequences specified by licensing bodies).
3. Mitigation is provided by the containment and associated systems.

2.3.2 Expanded safety strategy

The nuclear industry has, in general, achieved a high level of safety with the nuclear power plants (NPPs) operating today. However, it is the tendency of any industry to learn from experience and improve its product in the next generation. These improvements can result from many factors (e.g. operating experience, research results, new technologies becoming available) and are intended to provide a very high degree of safety and economic competitiveness.

During the last decade, in the aftermath of the TMI and Chernobyl accidents, the problems related to severe accidents - accidents beyond those considered to be design basis events - have been discussed quite frequently. Requirements for equipment specifically intended to minimize releases of radioactive material to the environment in the event of a core melt accident have been introduced. Such equipment (e.g., containment pressure relief systems with high-efficiency filters) has been installed in several NPPs, and additional plants are planning to do so. The trend for new designs is, however, to strengthen the confinement by increasing the design pressure of the containment so that venting will not be needed. The majority of designs for new plants include equipment for preventing or at least mitigating certain severe accident scenarios.

Many discussions, on both national and international levels, have concluded that the nuclear industries and regulatory authorities should try to establish a common strategy with respect to how the issue of severe accidents should be addressed for the design of future nuclear power plants. From a public acceptance point of view, this appears to be very important, in order to maintain nuclear power as a viable option for energy supply in the future.

The 1991 IAEA General Conference in its Resolution GC(XXXV/RES(para.9)) invited the Director General to start activities on developing safety principles for the design of future NPPs. As one step in this effort, in June 1995, the IAEA published the TECDOC-801 "Development of Safety Principles for the Design of Future Nuclear Power Plants". The TECDOC proposes updates to existing safety objectives and principles building closely on the INSAG-3 document "Basic Safety Principles for Nuclear Power Plants", Safety Series No. 75, and the INSAG-5 document "The Safety of Nuclear Power", Safety Series No. 75. The objectives and principles proposed in TECDOC-801 reflect lessons learned from operational experience, research, design and analysis efforts completed to date as well as emerging technologies. TECDOC-801 is intended to be useful to the designers, owners, operators, researchers and regulators involved with future NPPs as they develop and implement safety principles applicable to their specific designs and national situations. Implementation of these principles should lead to reactor designs with a very high degree of safety, recognizing that as a practical matter these future designs must also result in economically competitive energy production if they are to be utilized.

A key proposal of TECDOC-801 is that severe accidents beyond the existing design basis be systematically considered, and explicitly addressed if appropriate, during the design process for future reactors. Severe accidents are addressed by a **Technical Safety Objective** which is in part "..... to ensure that for all severe accidents addressed in the design there are no serious radiological consequences; and to ensure that the likelihood of any severe accident that could have serious radiological consequences is extremely small." The focus here is on protecting public health and safety, including explicit consideration of certain severe accident sequences. Severe accidents are further treated by a **Complementary Design Objective**: "To ensure, in addition to meeting the Technical Safety Objective, that severe accidents addressed in the design have no significant radiological consequences." This **Complementary Design Objective** accommodates the desire of many countries to demonstrate that no significant radiological consequences would occur outside the immediate vicinity of the plant, and thus no stringent off-site emergency response actions (such as prompt notification and/or evacuation, resettlement, etc.) would technically be necessary, for designs that meet this objective.

As noted in the TECDOC, not all conceivable severe accidents need to be considered, and if necessary addressed in the design. The selection of severe accidents to be considered is to be based on a combination of best estimate deterministic analyses, probabilistic considerations, including the application of numerical safety targets as guidelines, and a certain amount of engineering judgement. It is recognized that as a result of this process, a decision will be made to exclude some severe accidents of extremely remote likelihood from the set of severe accidents to be explicitly addressed in the design. Reaching this final decision is an iterative process, with initial judgements made by the designer, based on experience, and careful analysis, followed by review by utilities and regulators to confirm that the proper decisions have been made; these reviews may then result in further requirements on the design.

In deciding on the design features to be provided in future plants for coping with severe accidents, both the prevention of core damage and mitigation of its consequences are pursued. In selecting the means for addressing severe accidents, practical issues such as constructability, maintainability, and cost targets are taken into account.

Finally, and very importantly, TECDOC-801 notes that design features that are provided to address severe accidents are not expected to meet the same stringent design criteria and requirements (redundancy, diversity, and conservative analysis and acceptance criteria) applied to the engineered safety features to cope with design basis accidents. However, design features for addressing severe accidents are still to be engineered in a way which would give reasonable confidence that they are capable of achieving their design intent.

2.3.3 Extended design conditions

These new or extended safety objectives with the recommendations that the safety strategy for future plants should be expanded, over that for currently operating plants, by a new, fourth level, in addition to the

three levels described in Section 2.2.1, dealing with prevention and mitigation of "beyond design" accidents and conditions, are clearly reflected in the utility requirements documents, particularly in the EUR. Hence, it is obvious that they are, or will be, considered for all the advanced light water reactor designs that are currently under development.

To this end, the utilities behind the EPRI URD have introduced a "safety margin basis" concept that envelops and goes beyond the formal licensing basis requirements, thereby embedding more conservatism, and additional margins, in the design, and creating a capability to withstand also events/accidents beyond the current design basis. Designers and vendors have conducted research activities to investigate possible design measures to cope with a molten core in- and outside the reactor pressure vessel, and a number of solutions have been suggested. The regulators have in many countries been reluctant to formalize specific requirements with respect to "beyond design", or severe, accidents, but in some countries, e.g., Finland (part of the general design criteria), France, Germany (required by the Atomic Energy Act) and Sweden, requirements on adequate measures to cope with a molten core have been brought into the "licensing" basis. In some other countries, such as Japan, the regulators simply ask the applicant (designer/ vendor/utility) for his proposal to cope with severe accidents.

The implementation of the fourth level is made through the consideration of design extension conditions that encompass a number of situations that formerly were considered beyond design due to their low likelihood, and therefore not addressed. Typical situations would be complex scenarios involving combinations of initiating events and successive multiple failures of safety systems leading to core degradation situations and potential large releases of radioactive material. Design extension conditions would not require analyses in accordance with the rules and methodologies established for the design basis accidents; instead, they should be analyzed using best estimate analysis methods and sound engineering practices.

2.3.4 Identification of severe accidents to be addressed

Different approaches for identification of severe accidents for the design of future nuclear power plants were discussed at an IAEA Technical Committee Meeting held in Vienna in October 1995 on Identification of Severe Accidents for the Design of Future NPPs. It was observed that

- Nuclear power plants are traditionally designed to cope with accidents resulting from a set of design basis events, and this forms a foundation and a starting point for the identification of severe accidents. Both internal and external events should then be taken into account. (The extent to which external events beyond those considered as DBEs should be addressed, remains an open issue for further discussion.)
- Selection of conceivable severe accidents that should be addressed in the design can be made with probabilistic, deterministic and engineering judgement methods; combinations of probabilistic and deterministic methods are used in most approaches to the selection process, with varying balances between the two. Specific values for probabilistic cut-off limits to restrict the total number of initiators and sequences to be considered and adequate processes to determine such values are open to further discussion.
- A proper balance between prevention and mitigation should be maintained, however, priority should be given to prevention in particular for accident sequences with a potential for early containment failure. All operational states including full power operation, low power and shutdown modes should be considered. In addition, accident management, including the prevention and/or mitigation of the degradation of the core in the vessel, of vessel failure and of containment failure is important for future designs.
- Many organizations utilize core damage frequency and large release probability limits with typical values of 10^{-5}/reactor-year and 10^{-6}/reactor-year respectively. These values are not applied by all organizations, however; different countries and organizations have their own specific numerical goals and definitions for large release, etc.

A review of trends for water cooled reactors presently under development resulted in a set of severe accident phenomena and challenges to be considered, and addressed (prevented or mitigated), if appropriate,

in their design. In short, typical challenges that need to be considered at the design level are: high pressure melt ejection and direct containment heating; hydrogen production and combustion in the reactor pressure vessel and the containment; steam explosions in the reactor pressure vessel and the containment; core - concrete interaction in the containment; containment bypass and loss of long-term heat removal.

Depending on the design, phenomena related to such processes as reactivity transients, recriticality events, or missile generation might also need to be addressed.

2.3.5 Severe accident challenges

Severe accident challenges and strategies to deal with them were discussed at an IAEA Technical Committee Meeting in Vienna held in October 1996 on Impact of Severe Accidents on the Design and Layout of Future NPPs, and some of the observations are delineated below.

When considering accident sequences involving core melt, special attention is paid to prevention of an early rapid failure of the containment, with subsequent large releases to the environment and significant health risk for the public. To this end, all severe accident challenges with the potential for very energetic releases inside the containment are systematically considered, and addressed through appropriate preventive design provisions, when needed.

This does not mean that delayed containment failures are disregarded, however; challenges having the potential for leading to loss of containment integrity in the longer term are also taken into consideration. When appropriate, these challenges are addressed through a wider span of candidate solutions, including preventive or mitigative features, as well as accident management provisions, depending on the cost effectiveness of the countermeasures in each case.

A summary of challenges and candidate strategies to deal with them are presented in Table 2-II.

Table 2-II. Severe accident phenomena and challenges and candidate strategies for dealing with them

Phenomena and challenges	Associated candidate strategy
Phenomena associated with core melt at high pressure (HPCM)	Primary circuit depressurization system
Phenomena associated with core melt at low pressure, in the short term after vessel melt-through • loads on RPV support and cavity walls • thermal loads due to the ejected molten material (DCH = direct containment heating)	Prevent melt-through by in-vessel retention, minimize mechanical loads by providing adequate space around vessel, and thermal loads by providing large spreading area to improve coolability
Phenomena associated with hydrogen generation and combustion:	Provide adequate containment size and geometry, use igniters and recombiners, and/or containment inerting
Steam explosion: • in-vessel • ex-vessel	In-vessel steam explosions no major concern; ex-vessel steam explosions normally prevented by adopting post-flooding of collection pit.
Molten core - concrete interaction (MCCI) and possible basemat melt-through	Prevention through retention in vessel or mitigation through design features preventing interaction between corium and concrete
Mass and energy releases to the containment, causing: • high pressure • high temperature	Prevent by design features ensuring long-term heat removal
Containment by-pass	Prevention through rugged containment design, and reliable isolation system

The progression of an accident to a severe accident involving core damage will in most cases take some time and may be stopped by using existing safety systems in an abnormal way - in so-called accident management procedures. If the accident progression can not be stopped, the temperature of the fuel in the core will increase substantially, and eventually the core will start melting. The molten core material (corium) will collect in the bottom of the reactor pressure vessel, and after some time it may penetrate the vessel wall and flow out into the containment.

If adequate cooling of the vessel can be ensured, the vessel melt-through could be prevented, and this is the strategy adopted by a number of advanced designs under development; they aim at retaining the molten material inside the vessel by submerging the bottom part of the vessel into a large pool of water.

A vessel melt-through while the reactor is still under high pressure, could challenge the integrity of the containment, and design measures that ensure efficient and reliable depressurization of the reactor system are therefore included in the designs, initiated automatically or by accident management measures; such measures are in fact included also in the designs that aim at in-vessel retention of the molten material. The depressurization serves to prevent a high pressure melt ejection with its potential for causing an early failure of the containment.

If the situation were to degrade, after the reactor has been depressurized, there will be a melt-through at low pressure. The corium flows into the containment where it is collected and/or and spread into a dedicated area where it can be cooled.

The corium collecting pit is in most designs postulated to be dry (without water) in order to avoid the steam explosion that is assumed to occur if the hot corium were to fall into cold water, and that might jeopardize the integrity of the containment[2].

Further consequences of the core melt accident have to be covered by additional (safety) systems. Important phenomena to be controlled are the H_2 concentration in the containment due to oxidation of fuel cladding material, cooling of the molten material, and prevention of interference with the concrete. In any case, the radiological impact to the environment have to be kept at a very low level; the radioactive material released inside the containment during a postulated accident progression should be retained inside the containment by means of a rugged containment system.

[2] Recent experiments at Ispra, Italy, indicate that the probability of a steam explosion following an injection of corium into a pool of water is extremely small, very much smaller than the earlier predictions that were based on experiments using a melt of Al_2O_3, a material with properties that are quite different from UO_2.

CHAPTER 3. ALWR DEVELOPMENT ACTIVITIES

3.1 COUNTRIES WITH PROGRAMMES AND ACTIVITIES RELATED TO ADVANCED NUCLEAR POWER PLANT PROJECTS

3.1.1 Finland

3.1.1.1 Current nuclear power plant capacity installed

Finland has four nuclear power plants in operation, with a total capacity of 2 310 MWe. These plants comprise two BWR and two PWR units with a capacity of 1 420 and 890 MWe, respectively. The BWR plants, delivered by Asea Atom (now ABB Atom), Sweden, are located on the Olkiluoto island on the West coast of Finland and operated by Teollisuuden Voima Oy (TVO = Industrial Power Company Ltd), and the PWR plants, delivered by Technopromexport, Russia, are located outside Loviisa in the South-east region and operated by Imatran Voima Oy (IVO).

These four plants that were brought on line between 1977 and 1981 produce about one third of the electricity consumed in Finland, and they have operated very well, with load factors of around 90% for quite some years and an average annual occupational exposure per plant of less than 1 man Sv (100 man-rem) over the last 10 years; Finland has for many years been the leader among the nuclear countries in average load factor.

The Olkiluoto nuclear power plant units, Olkiluoto I and II, were built to achieve a net capacity of 660 MWe, but uprated to 710 MWe in 1981-83 when the operating experiences of the first years had shown that this capacity increase could be gained with quite small changes. With respect to contracting, Finland has experience from one total turnkey contract project (Olkiluoto I), one Split package contract (Olkiluoto II), and from a Multiple packages contracts project (Loviisa 1 and 2) with the utility IVO acting as Architect Engineer.

3.1.1.2 Operating experience

All four nuclear power plants in Finland have in general performed very well, and statistics on the performance of the nuclear power plants have for several consecutive years placed the Finnish plants among the top performers in the world. A summary of the load factors of the BWR and PWR plants during the period 1990-95 is presented in Figure 3.1-1. In particular for the two BWR plants, the results are indeed good, with an average load factor of 94%; over the last ten years, the average load factor of the two plants is slightly less, at 92.7%. [This is confirmed by the recordings from 1996 with a load factor of 92.5% for Olkiluoto I and 95.1% for Olkiluoto II, as reported by TVO.]

The radiation doses to workers and employees in the Finnish nuclear power plants have been among the lowest in the world. The average dose of all personnel exposed to radiation amounts to about 2 mSv a year, i.e., only 4% of the statutory annual limit of 50 mSv. The values of collective doses, including all outages, have also been low, with a ten year average per plant of 0.97 man Sv; annual plant occupational dose rates in excess of 2 man Sv were recorded twice during the period, at 2.05 and 2.25 man Sv, respectively, for Loviisa 1 and 2.

Finland is one of the first countries in the world with licensed and operating final repositories for low and medium level radioactive waste. TVO's final repository at the Olkiluoto site has been in use since 1992, and IVO is constructing a final repository on the island of Hästholmen, where the Loviisa nuclear power plant is situated. The IVO repository will be excavated 110 meters below grade in bedrock, and the waste facility is due to be completed by the end of 1996.

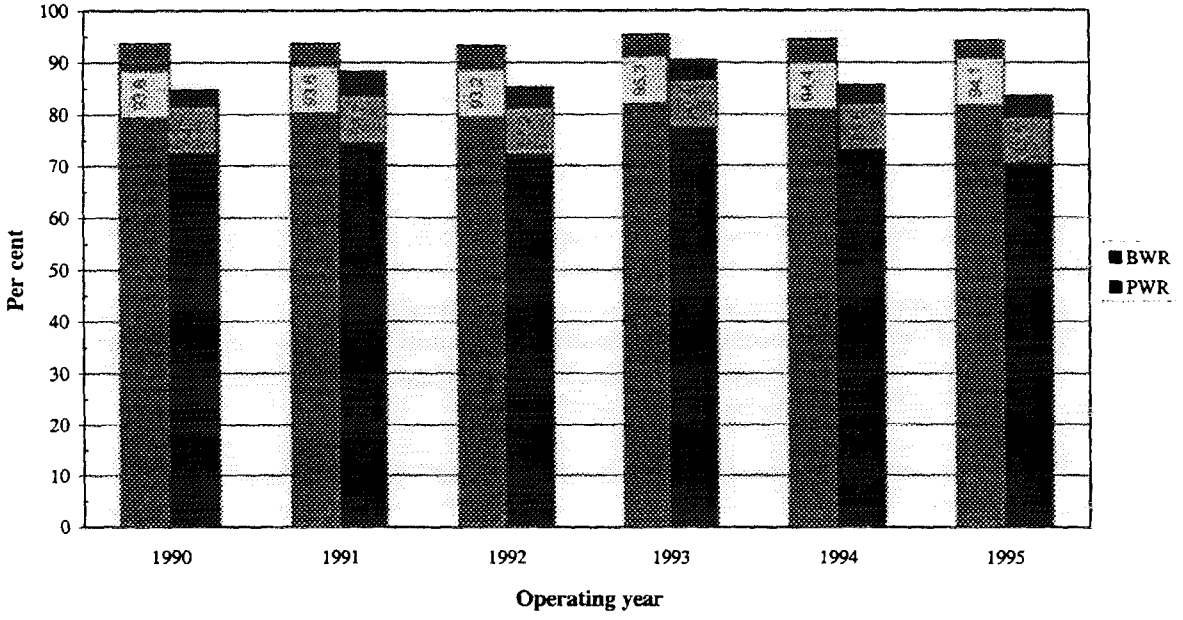


Figure 3.1-1 - Load factors of Finnish nuclear power plants

At the end of 1994, the Finnish Parliament ratified a change of the Nuclear Energy Act, banning all export and import of nuclear waste to and from Finland. As a result, IVO had to change its nuclear waste management policy; it will stop the spent fuel transports to Russia in 1996, and it has begun preparations for final disposal in Finland.

To this end, IVO and TVO have formed a joint company, Posiva Ltd., to research, build and operate a shared underground repository for their spent fuel. TVO has already been making investigations to characterize bedrock at three potential sites in Finland to define a suitable place for final disposal of spent fuel. The site for the repository is due to be chosen in the year 2000.

3.1.1.3 Nuclear power expansion plans

A complete revision of Finland's nuclear legislation from 1957 was finalized in 1988, marking a major reform in the public supervision of the use of nuclear energy. The new legislation will give society increased control over nuclear plant licensing and intensify the control of nuclear material and nuclear waste management. A key element of the reform is that decisions concerning the construction and operation of nuclear power plants and other nuclear facilities will now be made at the highest political levels, i.e., by the Council of State and Parliament.

In the years 1982-88, feasibility studies and development work were performed by IVO on the Soviet WWER-440 and WWER-1000 reactor designs, and the PWR design of Framatome was also studied. In parallel, TVO studied the ASEA-Atom BWR. In 1986, IVO and TVO established a joint company, the Perusvoima Oy (PEVO), to co-ordinate efforts aimed at construction of a fifth nuclear power plant unit in Finland. The purpose of this company was to enhance its owner companies' knowledge of nuclear energy, to study various reactor alternatives, and to follow international developments in the field of nuclear energy. The formal application for a design in principle (DIP) to an extension of Loviisa NPP or Olkiluoto NPP by a fifth unit was filed with the Government on 17 May, 1991 by IVO and TVO, and on 23 May, 1991, IVO, TVO and PEVO invited bids on the basis of thorough feasibility studies from three supplier candidates. In February 1993, the Government submitted a positive DIP proposal to the Parliament which voted on it in September 1993. The result of the voting was negative, 90 votes in favour and 107 against, and the application for the DIP of the fifth nuclear unit in Finland was turned down.

Since then, the utilities have stated that they will not make any new application until there is a clear indication of political support in the Parliament. At the moment, the main expansion plans refer to an increase in the power level of the existing units. These plans involve some 250-300 MW total increase by the end of the century.

3.1.1.4 Development programme on advanced LWR

In Finland, there is an on-going co-ordinated research and development programme, that is mainly focusing on operating reactors; the "Reactor Safety Research Program for the years 1995-1998". This programme is partly financed by the Finnish MIT (Ministry of Trade and Industry) which set up the Steering Group for it.

On the international scene, IVO Power Engineering Ltd (formerly IVO International Ltd) has acted as a consultant for Russian companies in the development of the WWER-1000 project for China (the WWER-91 type, or V-428 version). In addition, it is working, or has worked, in a number of TACIS PHARE projects for the Eastern Europe countries.

With respect to development of advanced LWR designs, Finland is participating in the development of the European Utility Requirements (EUR) for LWR plants. IVO Power Engineering is a participant in the European Passive Plant (EPP) programme that aims at adapting the Westinghouse AP-600 design to European conditions. In addition, it is following the european pressurized water reactor (EPR) programme. TVO is participating in the development work on the ABB Atom BWR 90 and in the European Simplified BWR (ESBWR) programme that aims at adapting the General Electric SBWR to European conditions, and it is participating in the development of the EUR, in particular with respect to BWR plant requirements (e.g., in the development of a Volume 3 for the ABB Atom BWR 90).

The requirements for a new nuclear plant in Finland consist of external or grid requirements, owners' requirements and licensing requirements. The grid requirements originate from the inter-Scandinavian or NORDEL requirements. The NORDEL grid has an installed capacity of approximately 50 GW, and the grid requirements are not expected to change appreciably in the near term future. With respect to plant sizes, the grid can accommodate full size units (1000-1400 MWe), and taking into account the economics of scale and the severe competition in kW h price, any new future plant is likely to be a full size unit.

The basic technical and economic requirements of the utilities (the owners) in the last bidding round were:

- The plant must be of proven design, i.e., of evolutionary type;
- It must be economically superior over fossil power during a 25 year calculated life;
- The construction period must not exceed 7.5 years;
- The technical service life time of 40 years shall be extended; non-replaceable components like the reactor pressure vessel shall be designed for 60 years of operation;
- The average life time load factor shall be 85%, and normal annual load factor 90%; and
- The annual occupational exposure shall be less than 0,5 man Sv (50 manrem).

With respect to the licensing issues, there are some areas where the plants offered at present would have difficulties to fulfill the Finnish requirements. These are: treatment of severe accidents including core melt; licensing of digitized instrumentation and control systems; and, as a minor point, applying the Leak-before-Break (LBB) principle.

3.1.1.5 ALWR licensing process

The legislation includes modern Finnish General Design Criteria, a set of Finnish safety guides, and a well defined licensing procedure which will apply for licensing of new, future nuclear power plants in Finland.

3.1.2 France

3.1.2.1 Current nuclear power plant capacity installed

France has 57 nuclear power plants in operation, with a total capacity of 59 948 MWe. There are two liquid metal cooled breeder reactors, the 233 MWe Phenix at Marcoule and the 1200 MWe Super-Phenix at Creys-Malville, on the banks of the Rhone river, with a total capacity of 1433 MWe. The other 55 French units are PWRs of various generations, with a total capacity of 58 515 MWe, for which the NSSS was supplied by Framatome and with Electricité de France (EdF) acting as architect engineer.

The policy of EdF with respect to nuclear has been standardization. The EdF units can be categorized into three series, namely 3-loop "900 MWe" units, 4-loop "1300 MWe" units and 4-loop "1450 MWe" units, the latter designed and built outside the original license agreement with Westinghouse.

Currently, thirty-four (34) "900 MWe" units, and twenty (20) "1300 MWe" units are in commercial operation, and the 1455 MWe Chooz B1 unit; the first unit of the N4 design, is undergoing commissioning tests. Three other units of the series are under construction, and they will start operation in the next years.

3.1.2.2 Operating experience

In 1995, French nuclear power plants generated 358.6 TWh, representing 76.1% of the total electricity production of the country. All French PWR units have performed well since they went into commercial operation, and currently, nuclear accounts for more than 80% of EdF's electricity generation. The time availability has been 79.2% for the 900 MWe units and 76.2% for the 1300 MWe units. These availability figures reflect the influence of long outage periods for 10-year overhaul or other heavy maintenance operations such as steam generator replacement. The figures for the last twelve (12) month period give a more representative picture of the unit availabilities, i.e., 83.4% for the 900 MWe and 84.3% for the 1300 MWe units. Average service time factors for the period 1990 through 1995 are presented in Figure 3.1-2.

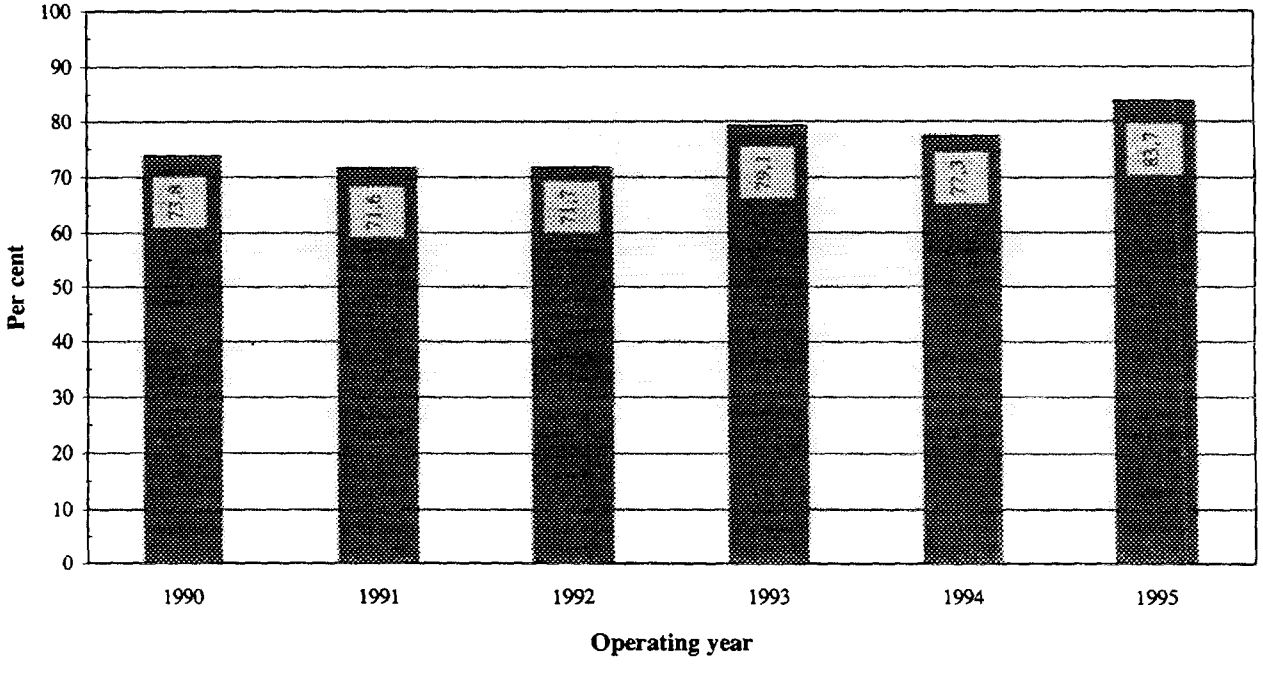


Figure 3.1-2 - Service time factors of French PWR power plants

After peaking at the end of the eighties, due to extensive ten year overhauls on 900 MWe units, the collective dose has been steadily decreasing, and last year the average exposure to EdF and contractor personnel was 1.63 man Sv. Discrepancies exist between the series, - the 1995 values for the 900 MWe and 1300 MWe plants were 1.96 and 1.07 man Sv, respectively, - and between the units in the series.

3.1.2.3 Nuclear power expansion plans

Three (3) units of the 1450 MWe N4 series are under construction, at Chooz and Civaux, and should be commissioned in 1997 and 1998. One of them, Chooz B2, on the Meuse river, should be commissioned in early 1997, and the other two, Civaux 1 and 2, on the Vienne river, should be commissioned in 1997 and 1998, respectively.

Considering the potential economic growth in the next few years, there is no need for new nuclear units on the grid, beyond the Civaux units, until the turn of the century. Developments underway are devoted to the definition of Utility Requirements which would be common to european utilities (EUR) and development of a new generation of nuclear units through the involvement in the design of the european pressurized water reactor (EPR) being developed jointly by French and German vendors and utilities. The EPR is scheduled to be commercially available by the turn of the century and might be chosen by EdF for a new series of standard units after the N4 series to be built and connected to the grid in the first decade of the next century.

3.1.2.4 Development programme on advanced LWR

Building upon the experience gained from operating a huge park of nuclear units, EdF decided to participate in the common work of a group of European Utilities, EUR, aiming at defining Utility Requirements for next generation nuclear power plants. The major objective of this document is to provide a common technical basis for utilities contemplating to order nuclear units at the beginning of the next century and thus prevent ratcheting requirements. Revision B of Volumes 1 and 2, dealing with general policies and top level requirements has been released in early 1996. Volume 3, presenting detailed requirements for specific designs, is currently being written for the EPR, and Volume 4, devoted to the balance of plant, is in preparation.

In parallel, a common French German industrial development is underway through the EPR project. The conceptual design phase and the consolidation phase aiming at the definition of plant architecture and of the main systems and components were completed by the first quarter of 1995. The basic design phase is currently underway and will be completed by mid 1997.

EPR is being developed for full compliance with both the EUR and French German safety requirements otherwise issued by the French and German regulatory bodies. The design emphasizes both safety and cost effectiveness.

In dealing with safety, prevention and system rationalization are emphasized, taking full advantage of operating experience feedback. Severe accident issues are also taken into consideration at an early stage of the design to ensure that design measures will have benign consequences on the day-to-day operation.

Cost effectiveness is addressed through a high electrical power output, in the range of 1450 MWe, and the goals of high availability and low operation and maintenance costs.

An R&D programme is also funded to investigate new solutions as appropriate.

Finally, a technical survey of most major initiatives in the domain of ALWRs is also carried out to provide a basis for comparison with the internal developments described above.

3.1.2.5 ALWR licensing process

The Revision B of the EUR document has been released to the French Regulatory bodies for analysis and further discussion with the utility.

After issuance of the GPR - RSK "Proposal for a Common Safety Approach for Future PWRs", and as the basic design of the EPR progresses, periodic discussions between the industry and the French and German regulatory bodies are organized to assess compliance of EPR design options and solutions with the above mentioned recommendations.

3.1.3 Germany

3.1.3.1 Current nuclear power plant capacity installed

In Germany, 19 LWR plants are currently in operation, with a total capacity of 21 984 MWe. These 19 plants comprise 13 PWRs and 6 BWRs, with capacities of 15 367 MWe and 6 619 MWe, respectively. The PWR has become the preferred reactor type for electricity generation in Germany.

Concurrent with the increase in number of nuclear power plants (NPPs) the need of standardization became evident. Siemens-KWU initiated in 1980, together with the utilities, the TüV experts and the licensing authorities, the Konvoi concept. For a series of successive NPPs, the procedure provided for uniform planning using identical software and hardware for all site independent equipment.

As a consequence, construction schedules and costs could be stabilized, and the construction time reduced. The Konvoi plants went into operation between three and seven months ahead of schedule.

This generation of German PWRs already incorporated "advanced" technology achievements that compare favourably to the development targets specified for many advanced, or next generation, PWR plant designs in other countries.

3.1.3.2 Operating experience

The German NPPs have with one exception, the Gundremmingen BWR, been built on single contract. A general advantage of this approach is that a number of generic problems to a large extent could be eliminated.

The implementation of any new technology in a project may be expected to lead to need for updates, changes and modifications of systems and components, as operating experience is accumulated. The modifications of the steam generator design for the PWRs of the German programme constitute a good example of proper feedback of operating experience. Changes of a number of parameters - water chemistry, material composition and properties and design - have together brought about a very reliable concept which is being used for all newer PWR projects.

The operating experience of the German BWRs and PWRs, as regards annual service time factors, is illustrated by the service time record (Figure 3.1-3) of the last six years, from 1990 up through 1995. The lower figures shown for the BWR are due to certain material problems and politically motivated long shutdown periods. A detailed break down analysis of all events that have occurred during the operating periods of the individual plants plant is not justified in this document; here, it may just be concluded that the operating experience with the LWR designs developed in Germany has been good, and that the actual standard technology constitutes a good basis for the development and future application of the next generation of NPPs.

3.1.3.3 Nuclear power expansion plans

The strategy of the German utilities implies that there should be a spin off reserve of 25% in the total installed electricity generation capacity. The actual total installed electricity generating capacity amounts to approximately 117 000 MW, out of which nuclear power plants account for approximately 20%. There has been a significant decoupling of the traditional increase in power consumption with an increasing gross domestic product (GDP), and the energy consumption has remained rather stagnant for quite some time. The energy demand forecasts have been revised drastically and there is currently really no need for a grid expansion, neither for nuclear nor for conventional plants.

Taking into account that some of the older NPPs may be shut down at about the year 2000, there may be a need for 1-2 new nuclear units. The rational behind this could be the introduction of the new EPR and to keep the nuclear option viable, reflecting the goals from the consensus discussions between the various political parties.

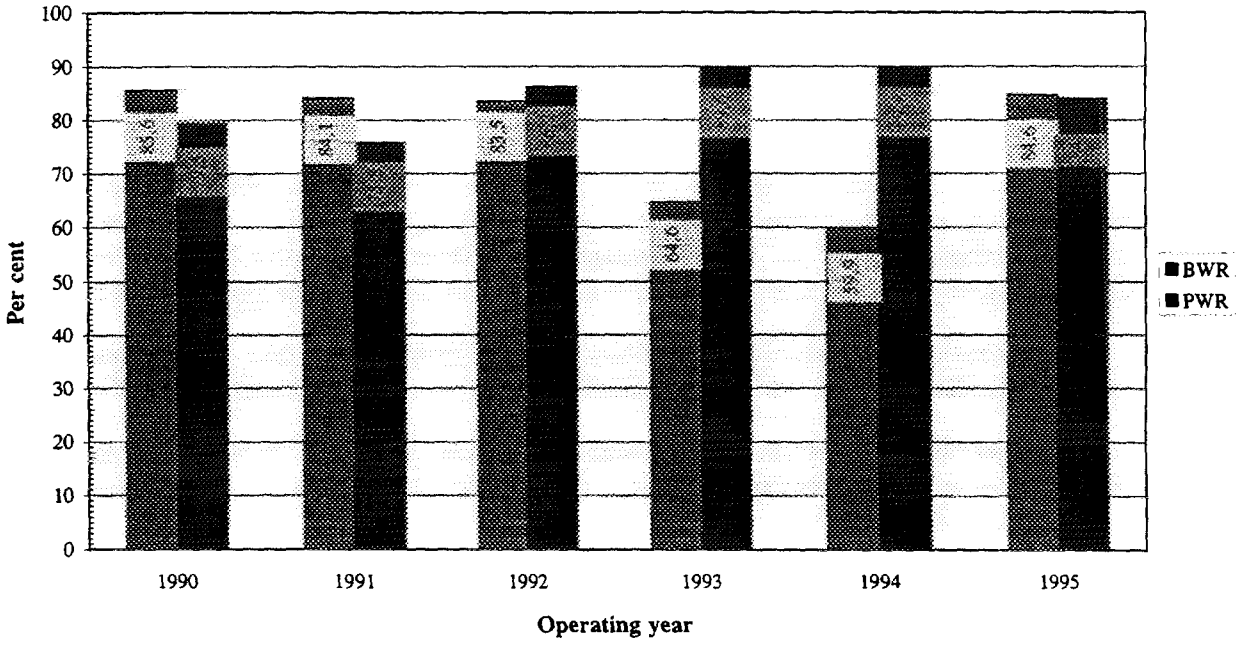


Figure 3.1-3 - Service time factors of German nuclear power plants

For Germany it must be concluded that no real grid expansion plans are currently considered by the German government and the utilities. However, certain development activities are under way, intended at not loosing the nuclear technology capability and to be ready to start building new NPPs at a certain time, hopefully not too far in the future, on the basis of the advanced technology that today is under development.

3.1.3.4 Development programme on advanced LWR

Based on the experience gained from the operation of the German PWRs, and in particular with the Konvoi plants, as well as from the huge PWR programme in France, the two countries have agreed upon a joint development programme for a PWR called the EPR. This development work was launched by Siemens, Framatome, and their jointly owned company NPI, together with the French utility EdF and a group of German utilities. The respective national authorities were also involved from the very beginning for solving the problem of licensing and code harmonization.

The conceptual phase lasted from 1989 to 1993, followed by a consolidation phase that ended with the start of the basic design work in the early 1995. The agreements on the basic design were signed in February 1995 and the work is scheduled to be ready by 1997. The technology of the new plant design reflects the decision of designing the plant on the basis of an evolutionary approach; in such an evolutionary approach the great experience of the existing plants can give confidence that the goals can be met.

The overall safety approach follows the proposal for a common safety approach for future PWRs issued by the French - German authorities. This requires the design to be made on a deterministic basis, supplemented by probabilistic analyses. Compared to existing plants, a twofold strategy is pursued. First, the preventive measures against accidents are to be improved. Second, even though the probability of severe accident scenarios - up to core melt - has been further reduced, additional features, mainly concerning the containment, shall be implemented to mitigate the consequences of such accidents.

For further improvement of the entire plant the following aspects are considered:

- Simplification of systems
- Use of proven technology
- Reducing the annual outage time
- Reduction of radioactive releases
- Increase of burn up of the fuel
- Increase in load factor;

all these aspects should lead to lower investment costs, lower fuel cycle costs and lower operating cost.

An important step with respect to improving the economy is the thermal power increase to 4250 MWth. Lots of optimizations are needed of several technical features which could be improved compared to current designs.

The German utilities and Siemens have launched also a further project, involving development of a 1000 MWe BWR with passive safety systems. The intent of this project is to establish an advanced BWR design that is based on the experience gained from current BWRs while taking into account new safety requirements, to be available for the utilities as an alternative to the PWR. Traditional safety systems will largely be replaced by passive systems to permit simplification of other systems and structures, in order to improve the economic viability; the economic viability compared with other energy sources must be granted by design.

The following development goals are considered:

- Achievement of a high safety level;
- Use of passive systems and inherent safety characteristics to cope with transients and accidents without interference by the operator(s);
- Consideration of severe accidents and their consequences (probabilities as low as for EPR);
- Reduction of the probability of occurrence of an unconfined accident;
- Design against external events (SSE of 0,25 g, protection against air plane crash (as for the EPR), and an explosion pressure wave with a peak value of 0,1 bar after 0,3 s);
- In the event of an accident, counter measures and control shall be by passive safety features without operator interference within 72 hours;
- Design for high availability and load follow capability (e.g., a single failure shall not lead to reactor scram);
- Increase of burnup capability of the fuel, and application of Mox fuel;
- Technical life time of 60 years; and
- Reduction of construction time.

The contract between Siemens and the German utilities runs for four years; during this period, the necessary R&D activities shall be completed, the Basic engineering work performed, and licensing documents prepared. The programme covers a BWR plant of about 1000 MWe that shall be licensable in Germany in accordance with German rules, regulations and requirements and meet the European Utility Requirements; the 1000 MWe size was selected to facilitate the adaptation into smaller and medium size grid systems.

From the R&D activities for this project, it can be noted that the test results of the emergency condenser have confirmed the theoretical calculations; the full cooling capacity was attained earlier than expected and margins in the capability of the system could be quantified..

3.1.3.5 ALWR licensing process

Together with other European utilities, the French and German utilities are preparing a document that addresses their future common design requirements for nuclear power plants. A first draft of this document, the European Utility Requirements, predominantly with specification for an advanced PWR,

was issued in April 1994; a revised document, considering comments given from vendors, followed in late 1995.

The French and German safety authorities have set up a joint committee to enable harmonization of the regulatory requirements. By mid 1993, general requirements were officially issued in a report entitled GPR - RSK "Proposal for a Common Safety Approach for Future PWRs".

A harmonization of the conceptual design features by the EPR partners was accomplished by mid 1993. This resulted in a comprehensive report describing the safety approach and the basic design objectives as well as the main technical features. The title of that report is: "Conceptual Safety Features Review File". This document was in September 1993 sent to the French and German licensing authorities for review of how far the French German requirements are met. Until the end of 1994, several key issues, e.g., APC protection, severe accident approach, radiological releases, have been evaluated and common recommendations have been issued by the safety authorities. Currently, discussions on the fulfillment of these recommendations are under way between the EPR planners and the authorities.

Furthermore, the authorities continue the evaluation/assessment process of further subjects. The report from the authorities is currently being updated with comments regarding the fulfillment of the established joint requirements, and the final report is due in the near future.

The acceptance of the proposed solutions in the field of safety, especially in the area of severe accidents, is an important step in the licensing process, and the early involvement of the respective authorities in the entire design process should eliminate the demands for major design changes.

3.1.4 Japan

3.1.4.1 Current nuclear power plant capacity installed

In Japan, 49 nuclear power plants are currently in operation, with a total generating capacity of close to 41,2 GWe (gross); the net capacity amounts to 39 499 MWe, according to IAEA PRIS data. These plants comprise 26 BWR and 22 PWR units with net capacities of 22 042 and 17 298 MWe, respectively, and one GCR with a net capacity of 159 MWe. The nuclear power plants account for approximately 30% of the total electric power output.

Four nuclear power plants have gone into commercial operation during the last two years (1994 and 1995):

- the Genkai unit no. 3 (Kyushu Electric Power Co.), a PWR of 1127 MWe net, in March 1994;
- the Kashiwazaki-Kariwa unit no. 4 (Tokyo Electric Power Co.), a BWR of 1067 MWe net, in August 1994;
- the Ikata unit no. 3 (Shikoko Electric Power Co.), a PWR of 846 MWe net, in December 1994; and
- the Onagawa unit no. 2 (Tohoku Electric Power Co.), a BWR of 796 MWe net, in July 1995.

Three units are under construction or being commissioned:

- the Kashiwazaki-Kariwa units no. 6 & 7 (Tokyo Electric Power Co.), two ABWR units of 1315 MWe net; unit no. 6 was connected to the grid in January 1996, and unit no. 7 followed in December 1996; and
- the Genkai unit no. 4 (Kyushu Electric Power Co.), a PWR of 1127 MWe net, in November 1996.

3.1.4.2 Operating experience

The first commercial LWR technology (in BWR and PWR plants) that was introduced in Japan, came from the United States of America. Troubles encountered in the early stages of adaptation of the technology resulted in rather poor operational records, but as a result of the "three stage improvement and standardization programs" organized by the Ministry of International Trade and Industry (MITI) the operational performance of the LWRs has been improved remarkably. Enhancement of reliability and safety, reduction of occupational radiation exposure and reduction of radioactive waste are the main achievements obtained from these programs.

The operating results of Japanese BWRs and PWRs with respect to annual load factors (utilization factors) for the six year time period 1990 through 1995 are shown in Figure 3.1-4 on next page. The annual load factors are generally above 70%, with an increasing trend, and the number of unplanned plant shutdowns have been at about 0.5 per reactor year for the past ten years. Consequently, it is concluded that the operating results achieved with the improved Japanese LWRs have been excellent, and this technology constitutes a good basis for the future development of the next generation LWRs.

3.1.4.3 Nuclear power expansion plans

The Japanese "Long Term Program for Research, Development and Utilization of Nuclear Energy" that was revised and published in 1994, indicates that the electricity generation by nuclear power is targeted at approximately 70.5 GWe and 100 GWe for the year 2010 and 2030, respectively.

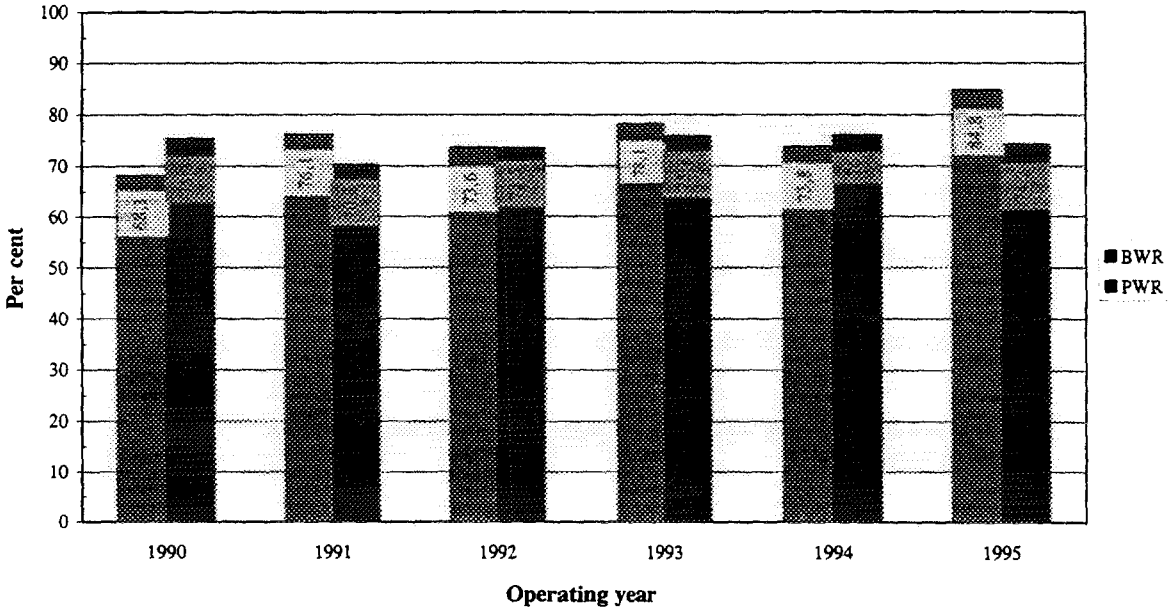


Figure 3.1-4 - Load factors of Japanese nuclear power plants

The nuclear power plant capacity has increased by some 1600 MWe per year through the period from 1970 through 1995, and the projections for the capacity increases up to year 2030 do not deviate much from this historical rate.

3.1.4.4 Development programme on advanced LWR

Advanced LWR plant designs (ABWR and APWR) were developed as the achievements of the three phases of the LWR improvement and standardization program were accumulated. After the basic development, the upgrading programmes of ABWR and APWR were continued, sponsored by Japanese utilities, and the Tokyo Electric Power Company ordered two ABWRs for construction at the Kashiwazaki site. Both these are now close to completion. No APWR plant has been ordered yet but the Japan Atomic Power Co. (JAPCO) has announced plans to build a twin unit at the Tsuruga site.

The studies and investigations for LWR designs to follow the ABWR and APWR are being carried out by MITI and by nuclear industry groups.

Studies by MITI

MITI has been making studies and investigations on how to proceed with development of future LWR designs intended for deployment between 2010 and 2030.

More specifically, the design philosophy of future LWRs is going to be formulated considering the social environment around Japan, enhancing safety and economic efficiency while also pursuing advancement of safety and regulation related to adopting the newest technologies, and taking into consideration international trends. Also, for the advancement of LWR technology, survey and review of nuclear technologies, domestic and foreign, that are in accordance with the design philosophy, are being performed.

The conceptual design of future LWR plants will be drawn up by considering the combination of new technologies and examining preliminary specification of systems and equipment and evaluating system performance, safety, reliability, maintainability and economics.

For the advancement of future LWR technologies, a "Working Group on the Prospects for the Future LWR Utilization" composed of people with experience from academia, electric utilities and vendors, etc. was formulated. Discussions of basic philosophy behind design objectives and plant concepts for the future LWRs were carried out in 1994 and 1995.

Studies by nuclear industry groups

An ABWR evolutionary programme was initiated in 1990, and at present, advanced technologies and plant concepts are being studied for the reference design with evaluations of economics and safety. The electrical output of the reference design is 1500 MW, but the possibility for an increase to 1700 MW is also being evaluated.

This study is being conducted as a joint study of Japanese BWR utilities and Hitachi, Toshiba and GE. One of the options being studied is adoption of large size fuel bundles to improve the core design and also shorten the fuel transfer time during annual inspection. Operating experience of the ABWR units at Kashiwazaki will be obtained in the near future and will be reflected in the development activity of the ABWR evolutionary programme.

In 1994, the PWR utilities, Mitsubishi and Westinghouse initiated a programme to develop the next generation PWR, the successor to the APWR. In this development, comparative studies of active and passive safety systems, as well as combinations of active and passive systems are being performed. Design requirements and conceptual design as well as research and development of applicable component technologies will be studied in the programme.

The new safety concept of the hybrid safety system which is an optimum combination of active and passive safety systems has been developed for a 600 MWe plant designated MS-600 by Mitsubishi. This concept has been selected as a main candidate to be studied further for a power output increase to about 1500 MWe in the second phase of the next generation PWR programme during 1996 and 1997.

Two other programmes are also underway: the Japanese Simplified BWR (JSBWR) and the Japanese Simplified PWR (JSPWR). The former is a joint study of the BWR group (the utilities and vendors of the evolutionary ABWR programme), and the latter is performed by the PWR group (the utilities and vendors of the next generation PWR programme) together with EdF of France. These Japanese Simplified BWR and PWR programmes study the adaptation and application of the passive concepts developed for the SBWR of GE and the AP-600 of Westinghouse to larger size plants, taking into consideration also the Japanese requirements of high seismic resistance.

The studies for 1000 MWe plants have been completed, and the feasibility of larger size Simplified Plants under stricter seismic requirements have been well confirmed. It is reported that the operability and maintainability will be considerably improved for both the JSBWR and the JSPWR by adopting simple design concepts. Studies on enhanced economy designs and power upgrade to 1200 MWe are being performed as next step programmes.

3.1.4.5 ALWR licensing process

The licensing process for the Advanced LWR designs has not yet started, but regulatory procedures similar to the current ones will most likely be adopted.

3.1.5 People's Republic of China

3.1.5.1 Current nuclear power plant capacity installed

In the People's Republic of China, three LWR plant units are currently in operation, with a total capacity of 2167 MWe. They are all of PWR type.

The third Chinese nuclear power plant, the second unit of the Daya Bay nuclear power plant, 906 MWe PWR units imported from Western Europe, was connected to the grid in 1994, bringing the installed nuclear capacity of China to 2.1 GWe. The first nuclear power plant built in China is the 288 MWe Qinshan, a PWR unit of own design that was connected to the grid in 1991.

3.1.5.2 Operating experience

The operating experience of the Qinshan unit is good. The Daya Bay units in Guangdong were connected to the grid in 1993 and 1994. At present, nuclear power represents hardly more than 1% of the electricity generation capacity of China, which mainly consists of coal fired power plants.

The average load factors over the last years are indicated in Figure 3.1-5.

3.1.5.3 Nuclear power expansion plans

China has expanded its electricity generation capacity at an annual growth rate exceeding 10% during the last decade. Up to year 2000 and beyond, China alone is expected to account for 20% of the increase in world electricity consumption.

In order to meet this rapidly increasing demand, China should build additional capacity at a rate of some 10-15 GWe per year during the next two decades. Coal is likely to remain the major electricity generation source in China which enjoys large domestic resources, and there is a potential for a substantial increase of the hydropower capacity since only 10% of the estimated reserves have been exploited.

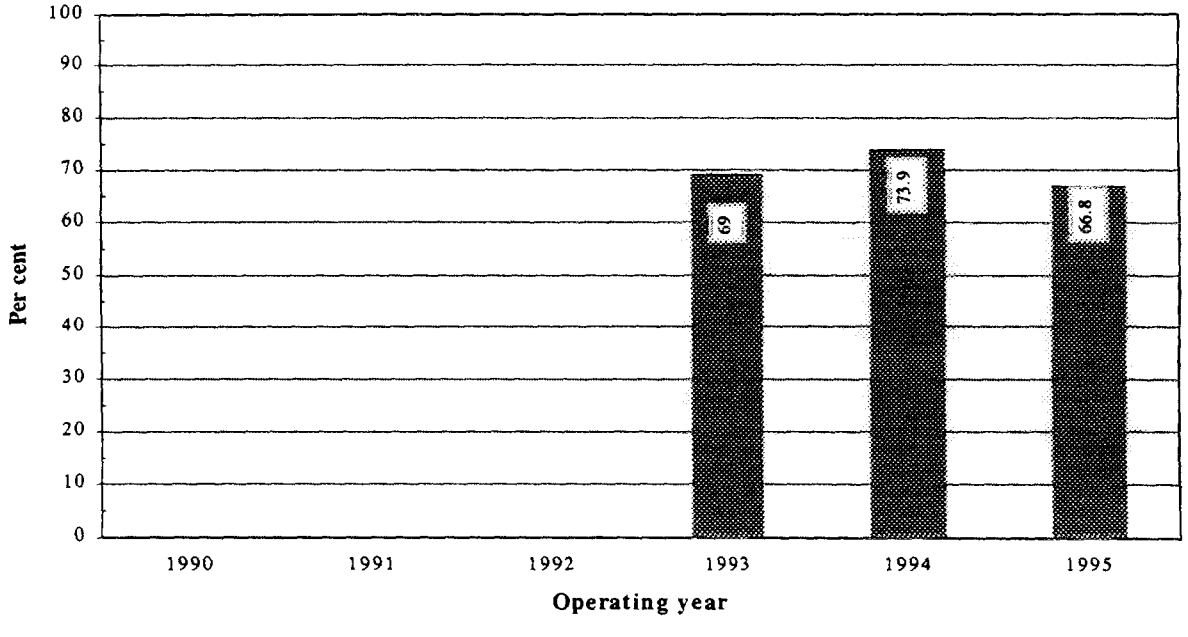


Figure 3.1-5 - Load factors of nuclear power plants in the Peoples Republic of China

For logistic, economic and environmental reasons nuclear power offers an attractive alternative option, however. About 80% of the coal reserves are concentrated in the north and northwest, and demand for more coal would lead to a large growth in transportation infrastructure, causing an environmental burden and an energy cost increase. Furthermore, coal burning is giving rise to concerns regarding atmospheric pollution, especially in densely populated urban areas.

The ambitious nuclear programme initiated by China aims at alleviating power shortages and environmental burdens. According to the China National Nuclear Corporation (CNNC), commissioning of more than twenty nuclear units, representing some 17 GWe, are planned for the next two decades.

A contract was signed in 1995 (in November) with the French vendor Framatome for the supply of two 900 MWe PWR units to be built at Lingao close to Daya Bay, scheduled for commissioning in 2002 and 2003. Another four to six 1000 MWe units are planned for the Guangdong Province.

At Qinshan, two 600 MWe PWR units of Chinese design are under construction and scheduled for commissioning in 2001 and 2002; these units are intended to be the first of a series of standardized units to be built at Qinshan and in other provinces of China.

In Jiangsu Province, the construction of two 1000 MWe units of the WWER-1000 type (of the V-428 version), imported from the Russian Federation, is planned, and design studies have started in co-operation with the Russian supplier.

Agreements have also been signed in November 1996 between the CNNC and the Atomic Energy of Canada Ltd (AECL) for the construction of two 700 MWe PHWR CANDU-6 reactor units at Qinshan. Additional CANDU-6 units could be ordered before the turn of the century.

In Fujian Province, where large additional electricity generation capacity would be needed to meet the demand, the construction of nuclear units is estimated to be a viable option provided that funding mechanisms can be implemented. Nuclear reactors for district and process heat supply are also being developed. A 200 MWth low temperature heating reactor is planned at Daqing City in the north-east of China, and an experimental 10 MW gas cooled reactor (HTR) is currently under construction.

3.1.5.4 Development programme on advanced LWRs

Based on the experience gained from the operation of the 300 MWe Qinshan unit and the 900 MWe units at Daya Bay, and from the design of the enlarged 600 MWe units for Qinshan phase II, the AC-600 designers are trying to find design solutions that will improve the economy and safety, in particular through use of system simplification, introduction of passive safety features, and modular construction methodology.

In this context, it may be noted that Chinese engineers and designers attentively have followed the discussions in the industrialized countries on safety issues, and the proposals for remedies and counter-measures. The different safety issues are therefore taken into consideration in the design development, and the advantages and disadvantages of the various suggested design solutions are evaluated carefully to see whether they could be beneficial with respect to the AC-660 design objectives.

3.1.5.5 ALWR licensing process

No licensing processes for ALWR designs have started yet; licensing has up to now generally been based on the rules of the country of origin, with some expert support from the IAEA. Taking into account the experience that has been gained through the previous projects, it is anticipated that specific licensing rules for China will be developed, together with a set of utility requirements, within the next decades.

3.1.6 Republic of Korea

3.1.6.1 Current nuclear power plant capacity installed

The Republic of Korea has continuously developed nuclear power generation since the introduction of Korea's first nuclear unit in 1978. Currently, there are 11 water cooled reactor plants in operation, with a total net capacity of 9120 MWe. These plants comprise ten PWR units and one PHWR unit with net capacities of 8491 and 629 MWe, respectively. In addition, 4 PWRs of 1000 MWe size and 3 PHWRs of 700 MWe size are under construction.

The PWR has become the main reactor type in Korea, and a standardized PWR design was established in 1986 when units 3 and 4 at Yonggwang were ordered. Along with the standardization, many improvements have been achieved in construction, operation and maintenance as well as cost savings for constructions and operation of PWRs.

The Korean Standard Nuclear Power Plant (KSNPP) design is now applied to the construction of units 3 and 4 at Ulchin, and will be applied to subsequent PWRs to be constructed in Korea until the development of the Korean Next Generation Reactor has been completed

3.1.6.2 Operating experience

The total installed capacity of power generation in Korea was 28 749 MWe by the end of 1994, of which the capacity of nuclear power plants was 7 616 MWe. The nuclear power plants have with respect to capacity become the largest electric power source in Korea with a 26.5% share, while oil-fired plants have decreased rapidly from 65.5% in 1970 to 19.9% in 1994.

In this context it can be noted that during 1995, another two nuclear power plants, units No. 3 and 4 at Yonggwang, with a gross capacity of 1000 MWe each, were taken into operation; Yongwang unit No. 3 was in reality connected to the grid already late 1994.

Since the start of commercial operation of Kori unit No. 1, Korea has achieved a steady growth in nuclear power generation. The total power generation in 1994 amounted to 164 993 GWh of which 58 651 GWh, or 35.5%, were produced by nuclear power plants. The average load factor of the plants in operation has continuously been maintained above 80% since 1991, as shown by Figure 3.1-6 on next page; in 1993 the average load factor was recorded as 87.2%, in 1994 as 87.4%, and in 1995 as 87.3%.

3.1.6.3 Nuclear power expansion plans

A new long term power development plan was established in December 1995, for the 16 years' period from 1995 to 2010. According to this plan, the total capacity of all power generation facilities will be 79 550 MWe in 2010, with 26 330 MWe, or around 35%, of this capacity being contributed by nuclear power plants. This means that the total capacity of nuclear power facilities will triple over the 16 years period.

During this period, the Korean Standard Nuclear Power Plant (1000 MWe PWR) and the Korea Next Generation Reactor (1300 MWe PWR) will play a major role while the PHWR remains an alternative nuclear reactor type.

3.1.6.4 Development programme on advanced LWR

Korea launched a next generation reactor development project, the so called KNGR project, in 1992, that aims at having completed a detailed standard design by the year 2000.

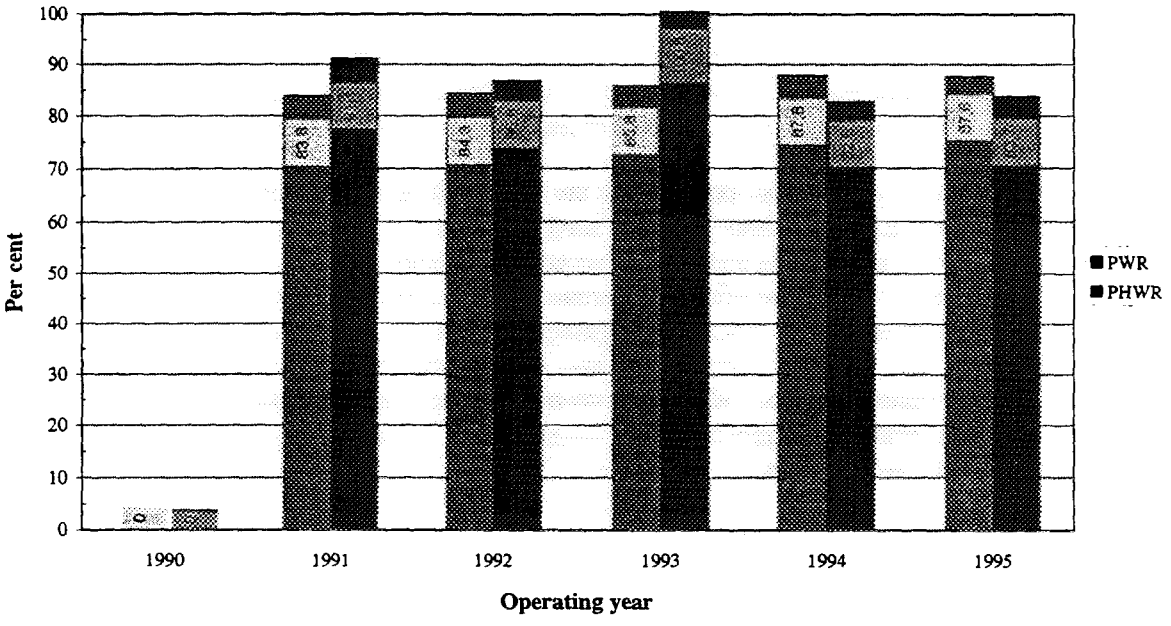


Figure 3.1-6 - Load factors of Korean nuclear power plants

The objective of this project is to develop an advanced design with a potential of becoming a safe, economical and environmentally sound energy source for Korea in the beginning of the next century.

For the development of the KNGR, an integrated project team that incorporates the specialized Institutes and companies of the Korean nuclear industry, has been set up. Korea Electric Power Corporation (KEPCO) is the leading organization in the project, and the Korea Atomic Energy Research Institute (KAERI) is responsible for part of the nuclear island design. Korea Power Engineering Company (KOPEC) is responsible for the overall nuclear island and turbine island design, Hanjung, formerly Korea Heavy Industry Company, is responsible for component design, and Korea Nuclear Fuel Company (KNFC) is responsible for the initial core design

The Center for Advanced Reactor Research (CARR), a university organization, is participating in the project to perform basic research activities, and the Korea Institute of Nuclear Safety (KINS), the regulatory IAEA, is also taking part, in order to enable early resolution of licensing issues.

The project consists of three phases of development activities from the start in 1992 to the completion in the year 2000. Phase I was a two year programme, running from 1992 through 1994, of which the major activities were the development of top tier design requirements and the design concept for the next generation nuclear power plant. Phase II is a three year programme, from 1995 through 1998, of which the major activities are related to the development of the basic design for the licensing review. Phase III, finally, will start in 1997 and run to the year 2000, and the major activity of this phase is the development of the detailed standard design to the level which will allow an accurate estimation of costs and construction schedule in order to ensure investment assurance.

The primary design requirements for the next generation reactor specify an evolutionary type pressurized water reactor (PWR) with a core thermal power of around 4000 MWt and a design life time of 60 years. The major considerations for this choice were: the level of maturity of the technology; the economy of scale; the availability of new nuclear sites; etc.

As safety goals for the KNGR, the requirements stipulates that the core damage frequency shall be less than 10^{-5} per reactor year and that the probability of core damage with subsequent containment failure shall be at least one order lower, i.e., below 10^{-6} per reactor year. These figures correspond with

the values given in the Utility Requirements documents developed by the Electric Power Research Institute (EPRI) of the USA (the EPRI URD) and the group of European utilities (EUR) and in recommendations from the International Atomic Energy Agency (IAEA); they really represent the current world wide consensus on what to aim at. In addition to the above probabilistic goals, limits to whole body doses and to the concentration of Cs_{137} at the site boundary have been established in order to provide technical soundness for a possible reduction of the emergency planning boundary in the future and to prevent land contamination in the event of design basis as well as beyond design basis accidents.

An economic goal has also been set for the KNGR; it should accomplish a cost advantage of at least 20% over electricity generation by a coal fired plant. In order to attain this goal, adaptation of new technology as well as introduction of advanced management systems will be necessary. The standardization policy and its benefits for operation and maintenance as well as spare parts inventory planning is emphasized in this context. Improved control and instrumentation systems, as well as improvements on man machine interface, are also seen as important in this context. As subsets of the requirements, the construction period is set at 48 months, modularized construction is high lighted, and development and systematic application of a reliability assurance program and a configuration management system are requested.

3.1.6.5 ALWR licensing process

The licensing process for the KNGR has not yet started, but the regulatory procedures are currently being reviewed; a licensing process similar to the one described in the 10 CFR.52 of the US Code of Federal Regulations is under investigation with respect to its appropriateness for the domestic circumstances.

3.1.7 Russian Federation

3.1.7.1 Current nuclear power plant capacity installed

In the Russian Federation, the nuclear power industry is considered to be a necessary component of the national fuel and power producing system, according to the national policy that has been developed for power production to meet the conditions of new economics in Russia.

At present, 29 reactor units of different types, located at 9 nuclear power stations, with a total generating capacity of 21 GWe (gross) are generating 12% of the total electric power; the net capacity amounts to 19 843 MWe, according to IAEA PRIS data. There are currently 13 LWR plants in operation, with a total capacity of 9064 MWe. These 13 plants are PWRs of the WWER-1000 and WWER-440 type. In addition, there are 15 LWGR (or RBMK) plants with a total capacity of 10 219 MWe, and one fast breeder reactor (FBR), the BN-600, of 560 MWe.

The currently operating WWER-1000 units are:

- the Balakovo units no 1 - 4;
- the Kalinin units no 1 & 2; and
- the Novovoronezh unit no. 5.

The currently operating WWER-440 units are:

- the Kola units no 1 - 4; and
- the Novovoronezh units no. 3 & 4.

The currently operating RBMK units are:

- the Bilibino units A, B, C and D;
- the Kursk units no 1 - 4; and
- the Leningrad units no 1 - 4; and
- the Smolensk units no 1 - 3.

3.1.7.2 Operating experience

The NPP operating experience of more than 7000 reactor years worldwide has demonstrated that nuclear power generation involves a minimum impact on the environment in comparison with fossil fuel power sources as long as no severe accidents occur. Therefore, prevention of severe accidents, and in particular such accidents that may be dangerous from a radiological point of view, is a key objective in the nuclear safety strategy.

The ensuring of safety of the first generation nuclear power plants (NPPs) is one of the most important tasks of the Russian nuclear power industry. In this respect, a variety of actions aiming at enhancement of safety and quality management of the operating NPPs have been performed during the last years.

The average energy availability factors of the Russian nuclear power plants have, as can be seen from Figure 3.1-7, been at a relatively low level the last years, but still rather satisfactory compared to the achievements of conventional fossil fired plants; the low factors are direct consequences of the transition of the Russian economy.

Examination of causes of unscheduled shutdowns and idle time periods has revealed that these were due to equipment failures, insufficient scientific and technical justification of design solutions, operating and maintenance personnel, and in particular operator errors.

A review of the status and operating experience of Russian NPPs, taking into account the priority of safety provisions for the operating NPPs, enables formulation of the following tasks:

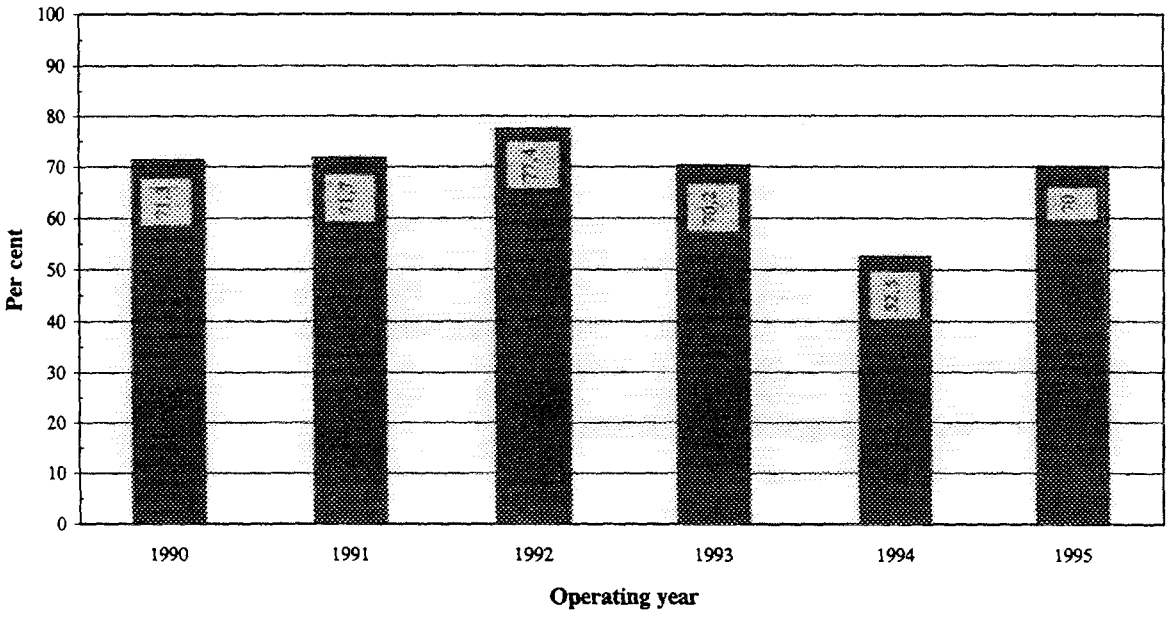


Figure 3.1-7 - Load factors of nuclear power plants in Russian Federation

- further work directed to ensuring safety of the first generation NPPs with WWER-440 units of the V-230 type;
- full scope implementation of the General Safety Enhancement Measures of the Operating RBMK and WWER nuclear power plants;
- development of methods and means for evaluation of the aging of NPP equipment; with improvement of maintenance routines and timely replacement of equipment based on these;
- improvement of the skill of the operating personnel; and
- improvement of the safety culture in the operation of nuclear power plants.

3.1.7.3 Nuclear power expansion plans

The existing fuel base of the nuclear power industry and the fissile material resources in Russia are large enough to support a considerable growth in nuclear power generation capacity.

With respect the subject of nuclear power expansion plans in Russia in the coming decades, two necessary stages in the further evolution of the Russian nuclear power can be mentioned. The first is a stage of renovation in which updating and safety enhancement of operating units would occur, whereas the second stage would exhibit a growth of installed capacity based on new generation power units (with enhanced safety).

The extent of nuclear power expansion within the framework of the Energy Production Strategy in Russia in the stage of the structural reformation for the coming 10-15 years is determined on the assumption of unconditional priority of energy saving policy and taking into account economic forecasts of the other constituents of the fuel and energy producing system. The "necessary" level of nuclear power by the year 2010 should provide 125 E9 kW h of energy production - with an installed capacity of 22 GWe. In this case, it should be borne in mind that NPP units with a total capacity of 8,4 GWe, will reach their design life time, and have to be replaced before the year 2010.

In addition to the scenarios stated in the Energy Production Strategy in Russia, expert evaluations of some other scenarios of expansion of the energy producing industry in Russia predict that an increase of the installed capacity by at least 25 GWe by the year 2015 is necessary.

Construction of new generation power units with WWER reactors will be prepared and started within the coming five years. WWER-640 pilot units of which design and parameters are based on national and world wide experience will be constructed at the Sosnovy Bor and Kola nuclear power station sites. WWER-1000 pilot units with enhanced safety will be constructed at the Novovoronezh site.

3.1.7.4 Development programme on advanced LWR

At present, work on the advanced WWER-1000 and WWER-640 designs is in progress based on the latest WWER-1000 design (the standard V-320 reactor plant design). The development of these designs follow two directions: the first is an evolutionary one, involving moderate design improvements and modifications, while the second is more innovative, incorporating implementation of new design solutions that increase the role of passive safety systems.

The evolutionary line aims at implementing design improvements based on:

- the experience gained from the operating nuclear power units; and
- the requirements of the current regulatory documents;

without a need for building and operating large scale prototypes. The NPP-91, a WWER-1000 design version with V-428 as type designation, is being developed jointly with IVO Power Engineering Ltd (formerly IVO International Ltd) of Finland, along the evolutionary line.

Some proposed design solutions for the advanced WWER-1000 and WWER-640 do require certain experimental, computational and theoretical justification work. For the WWER-640, these are related to confirmation of the design of the depressurization system of the reactor primary system, the passive residual heat removal system via the steam generators, and improved fuel with respect to failure rate, burnup, and cycle lengths. For the WWER-1000, such work is needed for the quick boron supply system and for passive heat removal systems.

For both designs, justification of the beyond design basis accident management procedures is an important task that requires performing and completing a large scope of research and development work.

Work is also going on at the Experimental Design Bureau "Mechanical Engineering", OKBM, on an advanced power reactor design in which the nuclear safety is ensured by large scale application of passive safety systems and inherent safety features. This design, the VPBER-600, is building closely on the principles developed for the nuclear heating reactor AST-500 that is under construction.

3.1.7.5 ALWR licensing process

Regulations and rules related to nuclear power plants and, to some extent, to research reactors have been developed in Russia, with the 1960s as starting point and with subsequent modifications. The current Russian regulatory requirements on nuclear power plants are, from the viewpoint of requirement scope and level, in general correspondence with the average world wide approach.

3.1.8 Sweden

3.1.8.1 Current nuclear power plant capacity installed

Sweden currently has 12 LWR plants, with a total net capacity of 10 005 MWe, in operation. These plants comprise nine BWR units with a capacity of 7300 MWe, and three PWR units with a capacity of 2705 MWe. The BWR units have been delivered by ABB Atom (formerly ASEA Atom), and the PWR units by Westinghouse. The output from these plants accounts for approximately 50% of the electricity generation in the country, with the other 50% predominantly coming from hydro power plants; other sources are contributing with only small portions of the electricity supply.

The nuclear power plants are basically of four generations. The first generation consists of the two first BWR plants - the Oskarshamn and Ringhals Units No 1, in operation since 1972 and 1976, - and the first PWR plant - the Ringhals Unit No 2, in operation since 1975. The three subsequent BWR plants - the Oskarshamn Unit No 2 and the Barsebeck Units No 1 and 2, in operation since 1975, 1975 and 1977, - represent the second. The third generation comprises two BWR and two PWR plants - the Forsmark Units No 1 and 2, in operation since 1980 and 1981, and the Ringhals Units No 3 and 4, in operation since 1981 and 1983, - and the fourth includes the two latest BWR plants - the third units at Forsmark and Oskarshamn, respectively, both taken into operation in 1985.

Most of the Swedish nuclear power plants have been uprated, i.e., the net output has been increased above the original rated power. The output of Barsebeck 1 and 2 has increased from 580 to 600 MWe (+3,5% from 1986); Forsmark 1 and 2 from 900 to 968/969 (+7,6% from 1989/90); Forsmark 3 from 1050 to 1158 MWe (+9,5% from 1991); Oskarshamn 2 from 580 to 605 MWe (+4,3% from 1988); Oskarshamn 3 from 1050 to 1160 (+10,5% from 1990); Ringhals 1 from 760 to 795 MWe (+4,6% from 1990); and Ringhals 2 from 820 to 875 MWe (+6,7% from 1991); - in most cases, with only minor supplements or modifications to the plant equipment.

3.1.8.2 Operating experience

The operating plants are characterized by high energy availability, short annual outages, and low occupational exposure. The good performance of both BWR and PWR plants during the period 1990-1995 is illustrated by Figure 3.1-8 in which the annual energy availability factors of BWRs and PWRs, respectively, are shown. The average annular occupational exposure for the BWR plants have been around 1 man Sv for a long period of years, but during the period 1990-95 it increased to 2 and even higher due to the repair activities in the generation 1 and 2 plants. For the PWRs, the development is encouraging; the average annual value has dropped from about 2 earlier to about 1 during the current period.

Detailed comments to different plant occurrences would take too much space, but a few observations affecting the comparison between BWRs and PWRs may be of interest. During restart of Barsebeck after the 1992 refuelling, a steam relief valve opened and substantial amounts of thermal insulation material were ripped off from the adjacent pipes. The insulation material fell into the condensation pool from which the core and containment spray systems draw water, and it partially blocked the suction strainers for these systems. As a result, all BWR plants with similar insulation material and pool arrangement were shut down - for repair, insulation material replacement and other modifications as needed to ensure that there would be no suction strainer clogging in the future. This shutdown order stopped Ringhals 1 and the generation 2 plants for about half a year; for Oskarshamn 1 it represented the start of a three and a half year shutdown. During the forced shutdown, cracks were discovered in its feedwater line nozzles inside the reactor pressure vessel, and OKG, the plant owner, decided to carry out a total verification of the reactor pressure vessel and other primary systems to check whether repairs, modifications and a restart could be justified on economical grounds.

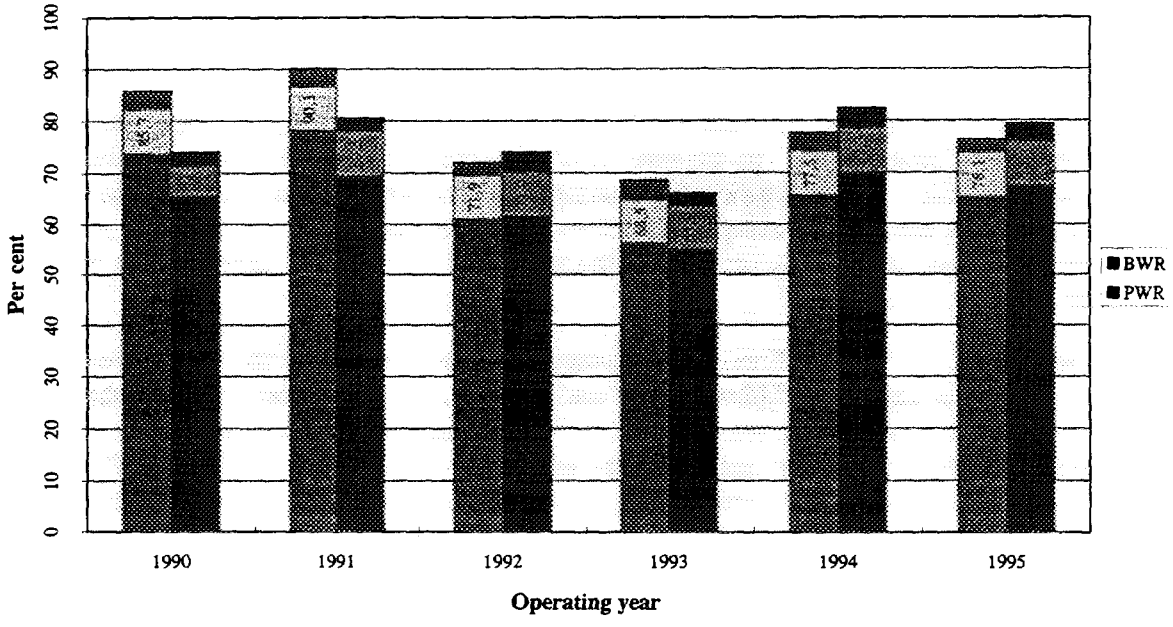


Figure 3.1-8 - Energy availability factors of Swedish nuclear power plants

For this verification to be made, a full system decontamination was carried out in January 1994, and about 99,5% of the radioactive material was removed, reducing the radioactivity level in certain parts of the reactor pressure vessel by a factor of 1000, from 20 to less than 0,02 mSv/h. The verification showed that after repairs of cracks etc. the plant would easily be capable of operation for another twenty years, and in January 1996, Oskarshamn 1 was again connected to the grid; a comprehensive plant modernization is also being planned.

Even the PWR plants had some problems; e.g., Ringhals 2 was down most of 1993 due to discovery of cracks in the reactor pressure vessel head, but generally, the PWRs operated well during the time period.

Systematic monitoring of the operating performance of the plants, evaluation of incidents and transients, and experience feedback are important for the industry, and for this purpose, ABB Atom and the nuclear utilities have formed a joint group, ERFATOM, to ensure that incidents and transients reported by the plants are assessed rapidly and carefully. There are also other areas of co-operation between the utilities and ABB Atom; present activities encompass systematic design reconstitution and modernization of the operating units, and programmes for experience feedback.

The design reconstitution programmes are broad in scope and involve several hundred man years of analysis and documentation. The safety case for the units is being reviewed, and their status in relation to new rules and guides examined to determine the need for modernization of plant procedures, structures, systems and components. These programmes are considered to be of paramount importance for an effective transfer of competence and nuclear know-how to the younger generations of engineers at the utilities and at ABB Atom, and the "Young Generation" initiative of ABB Atom and the utilities some years ago, now taken up also on the international level, also serve to strengthen this striving for "experience feedback".

The necessity of replacing "obsolete" equipment also contributes to initiation of modernization programmes. In particular, old instrumentation and control systems (I&C) are currently replaced by modern technology with programmable equipment. The replacement projects entail substantial modification of the I&C systems, including safety related logic circuits.

In the longer term, safety enhancement and modernization can pave the way for a further increase of the nominal power of the nuclear power plants; such further upratings are considered to be of interest from both economic and technical points of view.

3.1.8.3 Nuclear power expansion plans

In 1980, after the TMI accident, there was a referendum in Sweden with respect to the future of the nuclear power programme. The majority voted for a continuation of the programme and for keeping the plants in operation, provided their safety standard was acceptable, until alternative, renewable energy sources, providing energy at an acceptable price level, had been developed. In the wake of this referendum, the Parliament subsequently decided that the aim should be to have all nuclear power phased out by the year 2010.

At the end of 1995, an Energy Commission presented a report on the possibilities for and the consequences of a number of different phase out scenarios, and this report will form the basis for a political discussion during the Spring of 1997. The Commission did not advocate an early phase out, but indicated that a phase out of one unit before the turn of the century could be feasible. The cold winter weather during December 1995 and January 1996 showed quite clearly, however, that the system margins are far from adequate in such conditions, and that a phase out of nuclear capacity could jeopardize the electricity supply reliability.

The nuclear utilities which by law are forbidden to plan for construction of new nuclear plants in Sweden, is instead planning for modernizations and upratings of the existing plants; there is a wide spread feeling that even the politicians will realize that there are other, more important issues to spend national resources on. According to a public poll in March 1996, 65% of the population are now in favour of operating the nuclear power plants "as long as they meet the safety requirements" (without any specific end date), whereas 31% want an early phase out. [Still, in February 1997 the Social Democratic Government made an agreement with the Center party and the Communists to initiate the nuclear phase out by stopping one of the Barsebeck units on 1 July 1998.]

In this context, it should be noted that there is no urgent need for additional generating capacity; upgrading of the transmission lines to Denmark, Finland, Germany, and Norway may represent the best solution for the near term future.

3.1.8.4 Development programme on advanced LWR

As mentioned above, the LWR development in Sweden is not restricted to measures at the operating plants; it also involves new designs, even though there is a de facto moratorium on construction as well as planning of new nuclear power plants.

The Finnish utility Teollisuuden Voima Oy (TVO) operates two BWR units, Olkiluoto I and II, delivered by ABB Atom in 1979 and 1981, and talks about desirable technical features of a possible third unit at the Olkiluoto site were initiated later in the 1980s. In 1991, ABB Atom submitted a bid to TVO for such a third unit, as the fifth nuclear power plant in Finland. The proposed reactor was a 3800 MWt unit, designated BWR 90, based on the design of the most recently delivered 3300 MWt units delivered in Sweden, the third units at Forsmark and Oskarshamn.

Apart from the increased power rating, the development of the BWR 90 includes a series of improvements, such as reduction of building volumes and other measures to reduce cost, and introduction of digitized instrumentation and control systems (based on microcomputers). Other modifications involve the primary containment, primarily for improved mitigation of severe accidents.

In 1993, the Finnish Parliament voted against the construction of a fifth Finnish nuclear power plant, but ABB Atom and TVO have continued their joint efforts on further development of the design proposed in 1991, in a project denoted "BWR 90+". The following areas of review and development are typical examples on ongoing activities:

- Continued efforts to achieve short plant lead time and cost reduction during construction;
- Review of new licensing requirements and necessary design modifications for compliance, in particularly related to the regulatory guides (YVL) issued by STUK, the Finnish licensing authority;
- Review of the European Utility Requirements (EUR) and necessary adaptation for conformance; and
- Continued evaluation of items concerning severe accident phenomena and the adequacy of mitigation features.

Based on the experience from the operating plants in Finland and Sweden, very ambitious but potentially attainable goals have been set for the design and performance of the BWR 90+:

- Construction time, from first pouring of structural concrete to commercial operation: less than 54 months;
- The average capacity factor over 40 years operation; 85%
- Annual refuelling outage length: less than 20 days; and
- Annual occupational radiation exposure: not exceeding 0,5 man Sv (50 manrem).

The BWR 90+ represents a reactor type in the mainstream of evolutionary plants currently under development in the world, characterized by moderate modifications of existing reactor designs, taking advantage of the rapidly accumulating operating experience. A more "revolutionary" line of development has also been pursued by ABB Atom, the 600 MWe PIUS PWR plant design. In the PIUS concept, inherent safety characteristics and highly passive features and systems are employed consistently as "bottom" levels of defence against safety hazards. For lack of utility support, in Sweden and in other countries, further development activities at ABB Atom on the PIUS type of reactors have been put on hold; ideas based on the PIUS principle are being explored in several research laboratories in other countries, however.

Considerable interest, mainly in academic circles, is at present being displayed in the development of accelerator driven subcritical reactors. In these, accelerator generated protons are bombarding a heavy metal target, producing bursts of spallation neutrons which act as a source for the subcritical reactor. Such systems have been known since the 1950s and are now suggested to be suitable for the transmutation of actinides and long lived fission products into short lived nuclei. Concepts also exist for the development of accelerator driven systems for energy production. It is argued that the subcritical state of the reactor could bring about an enhanced level of safety, which the industrial community is quite skeptical to, but the technology still deserves some support as a research subject.

Concerted efforts are being made to ensure that competence in the field of nuclear technology is preserved and developed. The Young Generation initiative mentioned above is one example of this effort, and the Nuclear Technology Centre set up at the Royal Institute of Technology in Stockholm is another. The aim of the latter is to stimulate undergraduates and young engineers to take greater interest in nuclear matters, to pursue higher education leading to Ph.D. degrees in nuclear subjects, and to support academic research in subjects such as nuclear technology and safety, reactor physics, nuclear chemistry, and reactor materials.

3.1.8.5 ALWR licensing process

Specific ALWR licensing activities are not going on in Sweden, apart from following up developments abroad, since planning for construction of new nuclear power plants is not allowed.

3.1.9 United Kingdom

3.1.9.1 Current nuclear power plant capacity installed

In the United Kingdom, 35 nuclear power plant units are currently in operation, with a total capacity of 12 908 MWe. These 35 plants comprise 34 gas cooled reactors and one PWR, the 1188 MWe Sizewell B. The gas cooled reactors are of two different versions, the AGR (advanced gas cooled reactors) and the Magnox types (GCR). The gas cooled reactor fleet comprise 20 Magnox reactors in operation with a total capacity of 3360 MWe, and 14 AGRs with a total capacity of 8360 MWe.

British Energy operates seven (7) twin reactor AGRs plants and the PWR unit, with a total net capacity of 9548 MWe. Magnox Electric has six (6) twin reactor Magnox stations with a net capacity of 2960 MWe, and British Nuclear Fuels (BNFL) operate eight (8) Magnox reactors at Calder Hall and Chapel Cross with a net capacity of 400 MWe.

3.1.9.2 Operating experience

The operating experience of the United Kingdom's NPPs is illustrated by the service time record of the period 1990 through 1995 (Figure 3.1.9).

Sizewell B achieved commercial operation in July 1995, and its first planned outage for refuelling took place in June 1996. In the year 1 April 1995 to 31 March 1996 the plant delivered 7,9 TWh.

3.1.9.3 Nuclear power expansion plans

There are no immediate plans for further nuclear construction in the United Kingdom. British Energy intends to retain the option to construct further nuclear generating plants, but they do not expect that any decision to proceed with a new nuclear station will be made for a number of years. Replication of the Sizewell B design represents a possible option for future construction.

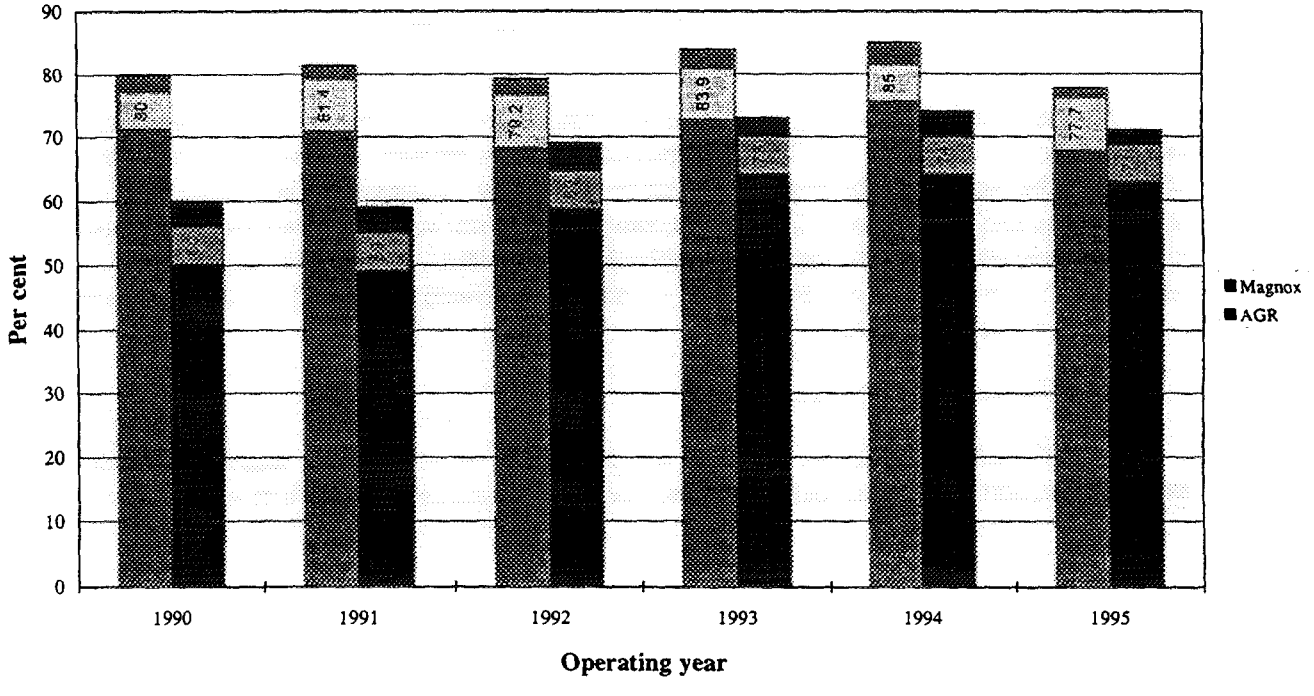


Figure 3.1.9-1 - Service time factors of nuclear power plants in the UK

3.1.9.4 Development programme on advanced LWR

The Sizewell B design represents a state of the art advanced light water reactor design and any future developments would be based on developing that design, particularly with regard to improving its economic performance.

The United Kingdom continues to participate in the development of European Utility Requirements through the EUR group and monitoring design developments taking place internationally.

3.1.9.5 ALWR licensing process

Together with other European utilities, the United Kingdom is involved in developing a set of common design requirements for nuclear power plants. Revision B of Volumes 1 and 2 of the European Utilities Requirements were published in November 1995. The licensing of future plants would take account of these requirements and licensing would be based on the safety assessment principles for nuclear plants published by the HSE in 1992.

3.1.10 United States of America

3.1.10.1 Current nuclear power plant capacity installed

In the USA, 110 LWR plants are currently in operation, with a total generating capacity of 102 580 MWe. These plants comprise 37 BWR and 73 PWR units with a capacity of 32 965 and 69 615 MWe, respectively. Since 1995, the number of nuclear power plants in commercial service had grown to 110 from 67 in 1980.

The US nuclear energy industry has been developing and improving light water reactor technology for more than four decades. Most of the plants built using this proven technology are virtually one-of-a-kind, custom designs. For future plants, the US government and the nuclear industry are firmly committed to developing standardized designs instead of the customized designs used in the past. These advanced reactors contain many features that make them even safer and more efficient than today's plants, and their standardized design will significantly reduce construction and operating costs.

Three new, standardized advanced light water plant designs are currently under development in the United States, and should be certified by the Nuclear Regulatory Commission in the 1996-98 time frame.

3.1.10.2 Operating experience

US nuclear power plant performance reached an all time high in 1995. In the United States, power plant performance is commonly measured by the capacity factor, which expresses a plant's actual annual output as a percentage of its maximum potential output. US nuclear power plants achieved a record high capacity factor of 78,8 percent in 1995, significantly higher than the 1980 average of 57,6 percent. Sixty-nine percent of the US nuclear plants (75 plants) operated at a capacity factor of 70 percent or better; just under one third of these plants achieved that level in 1980. Over fifty percent achieved a capacity factor of 80 percent or higher in 1995, compared with only 6 percent in 1980. Figure 3.1-10 depicts the trend in capacity factors since 1990.

Nationally, each percentage point increase in capacity factor is roughly equivalent to bringing another 1000 megawatts of generating capacity on line. Improved nuclear power plant performance thus helps meet the growing demand for electricity in the United States. In 1995, nuclear power plants provided 673 billion kilowatt hours (673 TWh), over one fifth of the total electricity generated in the United States.

The rise in capacity factor over the past decade is the result of plant modifications, improved operating and maintenance practices, and more attention to training of nuclear power plant personnel.

3.1.10.3. Nuclear power expansion plans

For future plants, the US government and electric utility industry is firmly committed to using standardized designs. The new designs will incorporate the latest technologies, and will be easier to operate and faster to build. These plants will achieve even higher safety goals than today's plants. Standardization simply means that all units of any type will be built in families of the same design, except for a limited number of site specific differences. Standardization will reduce construction and operating costs, and lead to greater efficiencies and simplicity in all aspects of nuclear plant operations, including safety, maintenance, training and spare parts procurement.

While final cost estimates for the new advanced standardized nuclear plants cannot be prepared until the Nuclear Regulatory Commission (NRC) approves the designs and the detailed engineering is completed, every effort is being made to ensure that these plants will be an economically competitive source of electricity.

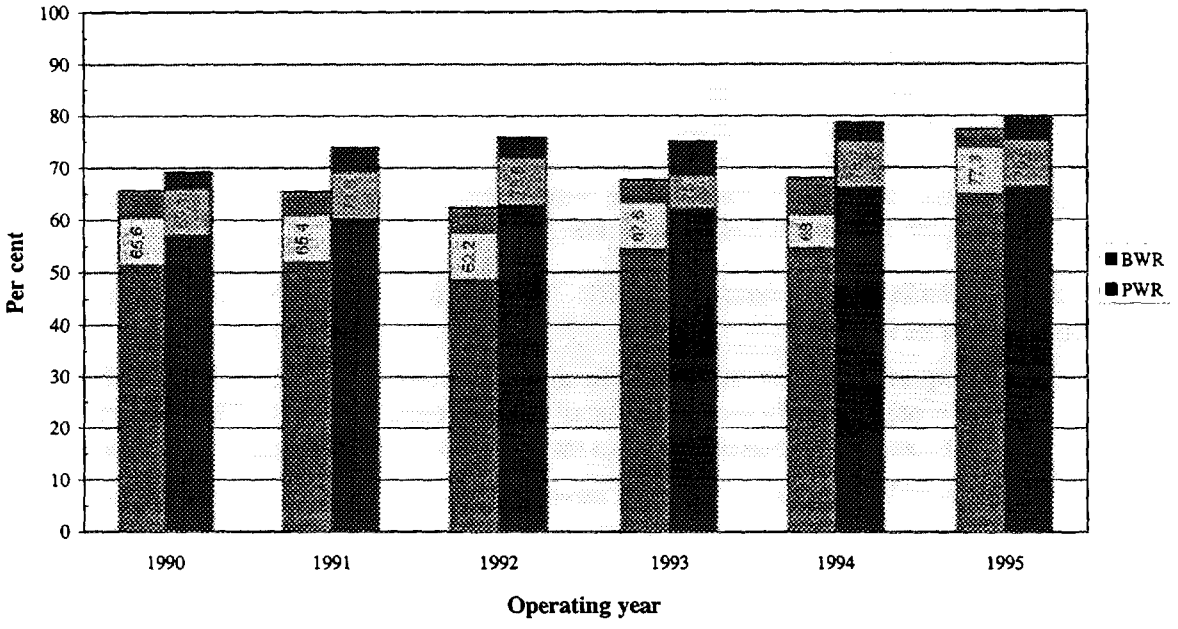


Figure 3.1-10 - Capacity factors of US nuclear power plants

Initial estimates for the cost of electricity from advanced light water reactors were prepared in mid 1992, based on assumptions that: the designs submitted to the NRC would be found generally acceptable; staffing levels and operating and maintenance costs for the plants could be optimized, based on the use of standardized designs, and would be at least as good as the best plants currently operating; and that, with the new licensing process and modular construction techniques, the plants could be built within four to five years after a combined construction and operating license is received from the NRC.

Given these assumptions, the analysis showed that standardized advanced nuclear power plants would be cost competitive with other types of electricity generating plants, including advanced pulverized coal plants, gasified coal combined cycle plants, and natural gas fired combined cycle combustion turbine plants.

Since the 1992 study, profound competitive pressures have begun to change the electricity generating business in the US As a result, the economic performance by advanced light water reactors will likely have to be even better than anticipated in the 1992 study. The reactor designers continue to refine and optimize the design, construction and operational aspects of these plants to ensure their cost competitiveness against other reasonable electricity generating alternatives.

Three standardized designs are under development in the United States. Two are large (1350 MW) "evolutionary" designs, and one is a smaller (600 MW) design that uses "passive" safety features. The passive design employs conventional reactor and power generation technology, but uses passive features such as stored water and gravity for safety functions as opposed to systems that use pumps and motors to move the water.

3.1.10.4 Development programme on advanced LWR

Two US nuclear companies - ABB Combustion Engineering Nuclear Systems and General Electric Co. - are designing large (1350 MW) light water reactors. They are called "evolutionary" designs because they build directly on the experience and lessons learned from plants already operating in the United States and around the world. Design and performance requirements were developed by the US government and the electric utility industry.

The evolutionary designs optimize the light water reactor, resulting in a plant that is simpler, easier to operate and maintain, and costs less to build. Safety studies indicate that these designs will be able to meet safety goals that are more than 100 times more stringent than those of current plants.

Since currently operating nuclear power plants were designed and built, there have been tremendous strides in many technological areas. Electronic cable systems are a good example. Today's nuclear power plants have miles of control cables. Tomorrow's plants will greatly reduce the amount of cabling required through the use of multiplexed, digital control systems, including state-of-the-art fiber optic technology. The new control systems are more reliable, more compact, easier to operate and thus safer. They are also simpler, which cuts construction time and cost.

Advanced Boiling Water Reactor (ABWR). The design for the ABWR differs from today's US BWRs in a number of ways:

- The safety improvements have resulted in a more compact design, and the ABWR's building volume is only about 70 percent of that of current operating BWRs. This will cut construction time and cost. It also makes the design more rugged and more immune to earthquakes.
- In the ABWR the control rods are driven by electro-hydraulic systems, as opposed to the simpler hydraulic design in current US BWRs. Having an additional drive mechanism reduces the probability of failure, and improves the plant's ability to produce electricity and to meet changes in electricity demand.
- All major equipment and components have been engineered with service and maintenance in mind, which will minimize downtime and reduce worker exposure to radiation.

Two ABWRs, which are very similar to the design being certified in the United States, have been built or are under construction in Japan. One unit began operation in January 1996 and the second is scheduled for 1997. The total construction schedule for these plants is about five years.

Advanced Pressurized Water Reactors. Like the advanced BWRs, the advanced PWRs are simpler designs. The volume of cooling water is greater, which will increase safety. They will operate at lower temperatures, and the "power density" of the fuel will be lower. The steam generator design incorporates advances in materials technology and has also been improved for ease of maintenance.

ABB Combustion Engineering is developing an advanced PWR, the System 80+. This plant is a further refinement on its System 80 design, now operating at several power stations in the United States and the Republic of Korea.

Through ABB CE's simplified design the construction manager can reduce construction time to only 48 months from first pouring of concrete to fuel loading. The System 80+ is engineered to achieve improvements in cost and safety with a number of significant features:

- Design margins are increased, reliability of existing systems is increased, and new safety systems are added.
- The control room and information processing systems have been totally revamped in ways that will reduce the burden on the operators and improve comprehension.
- The reactor is housed in a very large steel dual containment designed to withstand any credible accident.

Two ABB Combustion Engineering System 80 units under construction in the Republic of Korea incorporate many design features of the System 80+.

In 1984, a programme was initiated in the US to develop a new generation of mid size nuclear power plants (in the 600 megawatt range). The goal was to achieve even greater simplification in nuclear plant designs as a way of reducing cost and enhancing safety. Utility requirements were developed for

these "passive" designs, just as they were for the "evolutionary" designs. Westinghouse is developing a version of these smaller, simple designs, referred to as the AP-600.

Although this design is quite different from today's large reactors, it uses proven light water reactor technology and tested systems and components as much as possible. The concept relies on natural forces like convection and gravity flow of emergency cooling water, and thus minimizes reliance on pumps, valves, emergency diesel generators and other components that ensure safety in today's plants. In addition, the new design has improved automatic safety features.

There are several main differences between the mid size plants and today's nuclear plant designs. The AP-600 has several large tanks of emergency cooling water inside the containment structure above the reactor vessel. During an emergency, pressure and gravity would force this water into the vessel to cool the core. Compared to today's plants, the AP-600 will need 50 percent less building volume, 60 percent fewer valves, 75 percent less pipe, 35 percent fewer large pumps, and 80 percent less control cables.

Because of their simplicity, these smaller plants can be built much faster than recently completed US nuclear plants. Quick construction is possible because many systems and subsystems would be assembled as modules in the factory, not on the plant site. The goal is a construction time of three to four years.

3.1.10.5 ALWR licensing process

Both standardized 1 350 MW evolutionary plants have received final design approval from the NRC. The GE ABWR, in mid July 1994, became the first advanced reactor to receive its final design approval. The ABB Combustion Engineering System 80+ received its final design approval two weeks later, in late July. Certification of these two designs has been achieved in May 1997. Final design approval of the Westinghouse AP-600 is expected in the 1998 time frame.

The nuclear industry's *Strategic Plan for Building New Nuclear Power Plants* anticipates advanced light water reactor designs will be in commercial operation during the early part of next century. In February 1992, the Department of Energy and a consortium of electric utilities called the Advanced Reactor Corporation (ARC) signed a contract to launch a five year, cost shared, $276 million programme to do detailed, "first-of-a-kind engineering" on at least two standardized advanced light water reactor designs. This detailed programme completes most of the standardized engineering work that goes beyond what the NRC requires to conduct safety reviews, and will provide the level of information needed by potential buyers to estimate construction costs and schedules with a high degree of certainty.

In January 1993, two designs - GE's 1350 MW ABWR and Westinghouse's 600 MW AP-600 - were picked by the ARC to share first-of-a-kind engineering support. The first-of-a-kind engineering programme was completed for the ABWR in September 1996, and similar work on the AP-600 is underway with completion scheduled in 1998.

3.2 COUNTRIES WITH PROGRAMMES AND ACTIVITIES RELATED TO SPECIFIC DESIGN FEATURES OF ADVANCED NUCLEAR POWER PLANT PROJECTS

3.2.1 Belgium

3.2.1.1 Current nuclear power plant capacity installed

In Belgium, 7 LWR plants are currently in operation, at two sites, with a total net capacity of 5 693 MWe (per July 1996). These plants are all of the PWR type.

The country's first PWR reactor was an 11 MWe prototype unit that was started at the Mol Nuclear Research Centre in 1962 (the BR3 reactor); the first PWR built in Europe. This plant is currently being decommissioned. Apart from generating electricity this unit was used as a test bench for MOX fuel.

The seven operating units are basically of two generations. The first generation consists of the 400 MWe "Siamese" twin units Doel 1 & 2, in operation since 1974 and 1975, respectively, and the 980 MWe unit Tihange 1, in operation since 1975. The second generation comprises the last four units: Doel 3, with a net capacity of 970 MWe, and Tihange 2, with a net capacity of 930 MWe, in operation since 1982 and 1983, respectively, and Doel 4, with a net capacity of 1001 MWe, and Tihange 3, with a net capacity of 1015 MWe, in operation since 1985. These four units have a similar design, with a 3-loop NSSS, featuring a threefold safeguard system and an additional protection system especially designed to cope with external accidents.

All the plants are operated by ELECTRABEL, a private utility. Doel 1 and 2 are fully owned by ELECTRABEL, whereas the ownership of Doel 3 and 4 and Tihange 2 and 3 is shared between ELECTRABEL (96%) and the state owned utility SPE (4%). Tihange 1 is owned by SEMO, in which ELECTRABEL and Electricité de France (EdF) have equal shares.

All the stations were built using the multicomponent contract approach, enabling a high proportion of local manufacture. Overall some 90% of the investment was spent in Belgium.

No further units are currently under construction.

3.2.1.2 Operating experience

The nuclear power plants operate as base load units and have contributed significantly to the total electricity generation of Belgium. In the beginning of the 1990s, the nuclear share was about 60%, but the proportion has gradually decreased as new conventional power plants have been commissioned; in 1995 the nuclear share was about 55%.

All seven nuclear power plants in Belgium have always performed very well, with an average load factor of more than 80%, even in 1994 and 1995 when the steam generators were replaced at DOEL 3 and Tihange 1, respectively. (The DOEL 3 unit increased its net capacity by some 70 MWe with the replacement of its steam generators.) A summary of the average annual availability factors during the period 1990 to 1995 is shown in Figure 3.2-1. In 1995, the nuclear power plants produced 41.3 TWh.

From the beginning of February to the end of August 1996, a total of 27.4 TWh was produced. The average efficiency factor of the plants amounts to 82.1% in spite of a prolonged outage of Doel 4 for replacement of the steam generators, the condenser, and the process computer in the frame of the ten years safety reassessment of the plant. The total duration of these operations was 91 days and 7 hours with a collective occupational exposure due to the SG replacement of 0.63 man Sv.

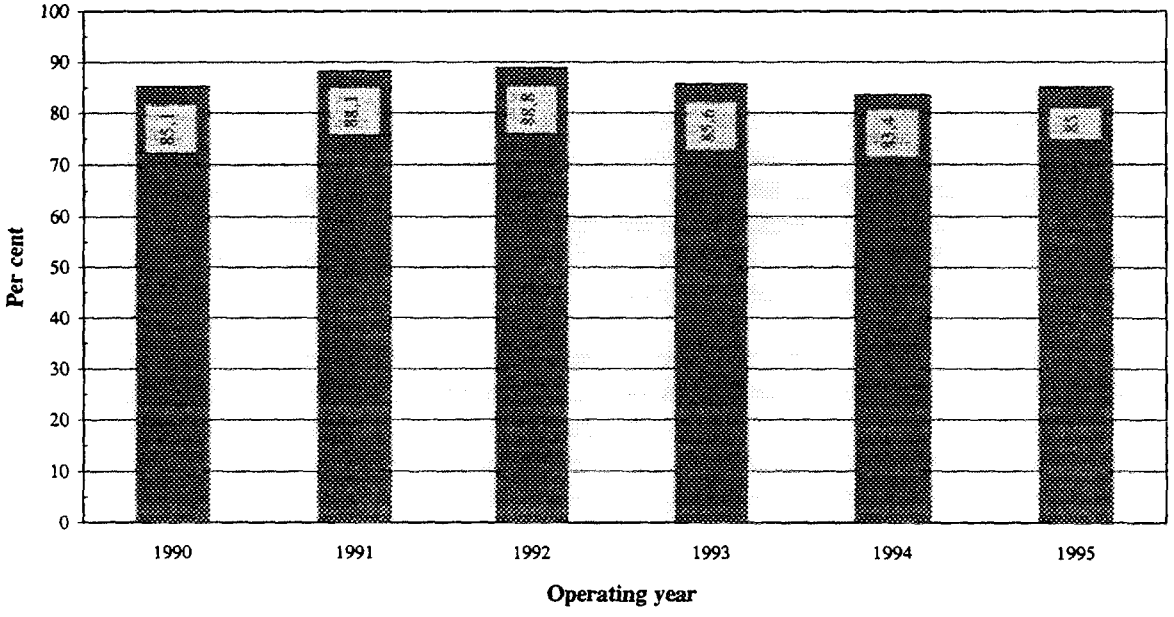


Figure 3.2-1 - Service time factors of nuclear power plants in Belgium

The replacement of the Tihange 3 steam generators is scheduled for 1998, but could be performed in 1997, if found required. The operation cycle length for Tihange 1, Tihange 3 and Doel 4 will be extended from 12 months to 18 months in 1998.

These satisfactory operating performances are due to the country's longstanding PWR experience. This includes the pursuit of high availability in all phases of design, construction and operation, the qualification of the domestic suppliers and labour and the training of operating personnel. Two simulator centres have been commissioned; one at Doel with a Doel 1/2 quasi full simulator and a Doel 4 full scope simulator, and one at Tihange with a Tihange 2 full scope simulator.

The first decision, made very early in the Belgian Nuclear Programme, addressed the protection of the public in the event of accidents. Therefore, a double containment system was engineered in order to ensure that the radiation dose outside the site would stay at approximately 0.1 Sv, even in very adverse situations (i.e., beyond design basis conditions, external accidents).

Interim storage facilities for spent fuel are added on each site. Due to site specific constraints, the wet storage option was selected for Tihange while the dry storage option was chosen for Doel. The water pool type facility of Tihange (for 3 700 assemblies) is presently under construction and will be operational in June 1997. The Doel power station additional spent fuel storage facility (for 2 200 assemblies) is an independent modular building for metallic, dual purpose casks (storage and transportation). The first module was commissioned by September 1995 and 6 casks are planned to be stored in this facility by the end of 1996.

Mixed oxide (MOX) fuel fabrication is routinely performed in Belgium by BELGONUCLEAIRE and FBFC International in their fuel fabrication facilities at Dessel. MOX fuel assemblies constitute 20% of the units fuel supply for Tihamge 2 and Doel 3, since February 1995 and June 1995, respectively.

A new facility (building 136) has been erected on the BELGOPROCESS Dessel site to store the conditioned high and medium level waste resulting from the Belgian spent fuel reprocessing operation at La Hague. In the facility area intended for the storage of medium and high level radioactive drums, the last equipment erection works are scheduled for the end of 1996. Final tests with the glass canisters transportation cask for vitrified waste were performed in September-October this year.

The state owned National Agency for Radioactive Waste and Fissile Materials Management (ONDRAF/NIRAS) is legally entrusted with the management and the final disposal of all radioactive wastes in Belgium. For the disposal of medium and high level long lived wastes, studies have concentrated on the use of deep clay formation and an experimental underground laboratory (HADES), the first of its kind in the world, was excavated and constructed 225 meters deep in the clay below the SCK/CEN site at Mol. The new PRACLAY project, jointly lead by ONDRAF/NIRAS and SCK/CEN, involves the feasibility demonstration programme for the final disposal of high level heat emitting waste in clay. The HADES facility extension work, i.e., the construction of a new access shaft and of a connecting gallery with the existing Test Drift, should start at the beginning of next year. The conceptual design studies of a facility for the packaging of the spent fuel assemblies in disposal containers are also in progress. ONDRAF/NIRAS hopes to present a preliminary safety assessment report on the concept by 2000, and the current target for the operating date is 2 035.

The dismantling of the second set of reactor internals of the BR-3 reactor was achieved in 1996. In total, the evacuated high level waste (i.e., with contact dose rate exceeding 0.2 Sv/h) amounts to about 13 tonnes containing 138 000 GBq of Co-60. In June 1996, it was decided to decommission the plant immediately and to continue the operation by first dismantling the remaining activated components, i.e., the reactor pressure vessel and the surrounding neutron shield tank.

Nuclear R&D in Belgium is mostly carried out by the CEN/SCK (Belgian Nuclear Research Centre) of Mol and is mainly confined to: reactor safety experiments, radioactive waste disposal, decommissioning, radiation protection and health physics. R&D to support NPP operation is carried out by LABORELEC, the central research laboratory of the utilities ELECTRABEL and SPE. The utilities are members of the Institute of Nuclear Power Operators (INPO), the World Association of Nuclear Operators (WANO), and the Westinghouse, Framatome and Steam Generators Owners' Groups.

3.2.1.3 Nuclear power expansion plans

No further nuclear power plants are currently being planned in Belgium. As a result of the Chernobyl accident, a governmental decision of December 1988 postponed the approval of the eighth nuclear unit (Doel 5) and precludes further investments in nuclear power in the country for the time being.

The electricity demand growth in Belgium is expected to remain low, at 1 to 2% per annum, but a considerable portion of the current conventional power plants, with an age of more than 25 years, are scheduled to be decommissioned in the next years.

The new 1995-2005 Belgian Equipment Plan confirms the 1988-1998 plan which forecasts the commissioning of combined gas turbine and coal fired power plants with a capacity of 3400 MWe until 2005. Additional nuclear capacity will appear as a result of the power upgrading achieved or to be achieved at the Doel 3 (106 MWe), Tihange 1 (106 MWe) and Tihange 2 (106 MWe) units. These power upratings arise from the improved efficiency of the new steam generators but also from the use of margins of the reactor core. Belgian utilities (ELECTRABEL 66.6% and SPE 33.3%) also hold a 25% share in the two French 1 450 MWe PWR units under construction at Chooz, close to the Belgian border, of which one is under commissioning and the other will be commissioned in early 1997. This implies an additional 725 MWe of nuclear capacity.

No new nuclear power plants will be developed under the existing 1995-2005 equipment plan. By the end of the century, the share of nuclear power in the total electricity generation is expected to decrease to about 50% (from the current share of about 56%). However, the nuclear option does remain open and there are plans to study the possibility of prolonging the operating life of the oldest power plants.

3.2.1.4 Development programme on advanced LWR

In the perspective of the construction of a new unit in the future, all the parties involved in the Belgian nuclear electric sector, in particular the utilities, their consulting engineers and the licensing autorities, follow closely the studies in progress regarding advanced light water reactors (LWRs).

For nuclear power to be a viable option for electricity generation in the future, nuclear power plants will have to meet two challenges. First, they will have to be safe and accepted as such by the public, and secondly, they have to be competitive compared to other means of electricity generation.

All over the globe, the nuclear industry (i.e., vendors, utilities and licensing bodies) is acting in closer concert to make available on the market a number of standardized products that have been approved in advance by the safety authorities. More specifically, the utilities operating NPPs have initiated programmes to develop requirements for the new generations of nuclear power plants that would have to be met by all vendors.

In this context, along with other European countries, Belgium participates in a number of international activities related to the development of advanced LWR designs:

- the Utility Requirement Document (URD) project of EPRI which following the setting up of the new requirements shifted to the design development and design certification of two large, evolutionary (ABWR and System 80+) and two "passive", mid size (AP-600 and SBWR) US reactor designs;
- the European Utility Requirements (EUR) project consisting of the preparation of utility requirements for future NPPs to be built in Europe; and
- the European Passive Plant (EPP) project which aims at adapting the Westinghouse AP-600 design to EUR requirements, including an upgrading to the 1000 MWe range.

With respect to the severe accident issues including core melt, Belgium is also involved in the ACE and RASPLAV projects. Severe accident procedures are presently being implemented in the operating power plants. passive autocatalytic recombiners (PARs) are installed inside the containment of the seven units to cope with the risk of hydrogen deflagration in the event of a core melt.

3.2.1.5 ALWR licensing process

US Rules and Regulations have been applied for the licensing of all current Belgian nuclear power plants, since national rules with respect to nuclear safety are limited to general licensing organization and to health physics standards.

The licensing procedure which takes place under the joint authority of the Minister of Labour and Employment and the Minister of Public Health and the Environment, is a one step procedure. The Ministers are assisted by a Special Commission, which includes people chosen for their qualification. AIB-Vincotte Nuclear (AVN), a state approved agency, carries out an official acceptance procedure for installations prior to the granting of the operation license.

For the time being, no changes are anticipated to the regulatory framework and the licensing process to be applied to advanced reactors. US Rules and Regulations will continue to be applied. However, if an European set of rules should be established and approved in co-operation with other European countries, then those rules could be considered by Belgium. Accordingly, the Revision B of the European Utility Requirements (EUR) has also been submitted to AVN with the objective of having them discussed and to possibly obtain an approval in co-ordination with other European Safety Authorities.

3.2.2 Italy

3.2.2.1 Current nuclear power plant capacity installed

In Italy, no nuclear power plants are currently in operation, following a referendum, and a subsequent Parliament decision, in the wake of the Chernobyl accident in 1986, to shut down the three nuclear power plants in operation (Caorso, Enrico Fermi and Latina) and to stop work on those under construction at Alto Lazio.

3.2.2.2 Operating experience

The 153 MWe gas cooled reactor Latina was shut down in 1987 after 24 years of operation, while Enrico Fermi (Trino), a 260 MWe PWR, and Caorso, a 860 MWe BWR, were shut down in 1990 after 25 and 12 years of operation, respectively.

ENEL was the operator of the plants, and it was also involved in constructing them, as the architect engineer. Hence, it can be noted that ENEL acquired considerable experience in constructing and operating nuclear power plants, and this know-how has been utilized for development programmes in other countries as described below.

3.2.2.3 Nuclear power expansion plans

As noted above, the Italian Parliament decided to shut down the three nuclear power plants that were in operation and to stop construction of two other units, and there are no expansion plans. The political decision to abandon nuclear power does not per se represent an absolute and everlasting moratorium, however; the politicians have left the door somewhat open for a new review after a decade, if new designs with significantly improved environmental protection were to become available.

3.2.2.4 Development programme on advanced LWR

In spite of the fact that no nuclear power plants are operating in Italy, the Nuclear Division of ENEL is fully involved in the development of nuclear power plants of advanced design in a wide international context since 1988. In this framework, ENEL has devoted a major effort to support the new objectives for environmental protection that aim at ensuring a minimum of environmental impact also in the event of severe accidents, thereby providing a technical basis for significant simplification of the emergency protective actions, avoiding evacuation, permanent relocation of the population and limiting land contamination. These new objectives that were early proposed by ENEL, can be recognized both in the Utility Requirements Document (URD) of the Electric Power Research Institute (EPRI) and in the European Utility Requirements (EUR), and they are extensively supported in the international nuclear community.

At the same time, the four most committed designers in developing a new generation of nuclear power plants in the world, viz. ABB, General Electric, NPI, and Westinghouse, with their new improved designs, are demonstrating that the new requirements can be met.

In addition to contributions to the efforts of setting up utility requirements for the next generation of nuclear power plants, ENEL has been involved in significant research programmes and studies, mainly through participation in development programmes supported by the utilities. Special attention has been dedicated to: the containment integrity, the analyses of reactivity transients, the study of coolability and confinement of the molten core, the analysis of accidents, the behaviour of radioactive steam and the decontamination of systems and apparatus.

The main ENEL involvement in R&D and development efforts in the nuclear field are summarized below.

The EPRI/DOE ALWR program

ENEL has been a member of the ALWR Program since 1988. At that time it was very active in writing the new requirements, and later on, the focus shifted to the development of the designs, in particular to the two simplified, passive plants: the SBWR of GE and the AP-600 of Westinghouse. For both designs, ENEL has performed significant parts of the Probabilistic Safety Assessment (PSA) and, in co-operation with other industrial partners, major tests.

The European Requirements and related designs

In 1993, ENEL joined a group of European utilities that was established to write the European Requirements for future reactors. This group comprised seven of the most important European utilities with an interest in nuclear power, and the first effort was devoted to developing the general requirements that will apply to all plants. These will be followed by special requirements for each standard plant under consideration. ENEL, in particular, has been responsible for comparing the EUR with the EPRI URD, and for developing important parts as safety aspects and containment.

This effort of electric operators has a pendant in an initiative to develop an evolutionary European pressurized water reactor (EPR). This effort will take advantage of the wide experience of design, construction and operation of French and German pressurized water reactors, in particular the N4 and Konvoi reactors, respectively, and also the operating experience of the most important European electric utilities. In accordance with the consolidated trends in Europe, the EPR will have a double containment, consisting of a primary containment in prestressed concrete surrounded by a secondary containment in reinforced concrete.

Another initiative was taken in 1994 related to developing an adaptation of the AP-600 complying with the European requirements. Eight electric utilities and Westinghouse agreed on a co-operation programme aimed at generating, first of all, a reference configuration (EPP) suitable for the European market. This reference configuration will be subjected to a general review by the Safety Authorities of the interested countries, among them Italy. ENEL was assigned the chairmanship of the EPP design development effort.

The Westinghouse AP-600 and EPP projects

The AP-600 design which is part of the ALWR development programme in the United States, introduces some remarkable innovative elements in the technology of pressurized water reactors (PWRs), in order to achieve essential improvements with respect to reliability, simplicity of operation and safety response, as well as a reduction in construction time and costs. The AP-600 plant design combines a few special design solutions, such as the innovative containment system, the adoption of passive safety systems, and a different role of the operator, with conventional design solutions taken from the current technology of PWR reactors. The innovative features of this reactor spurred ENEL to evaluate the feasibility of design modifications to ensure, as appropriate, conformity with specific Italian requirements. Since 1994, this effort has been redirected to the EPP programme which is a three phase programme. In the first, the adaptations needed to comply with the European utilities requirements for next generation nuclear power plants are being defined, and in the second, a reference design will be developed and documented in a safety report.

Independent studies of the containment

1. Concrete containment and passive containment cooling system

As stated above, ENEL started studies on modification of the AP-600 containment system that could meet Italian criteria while observing the European trend. This modification involves using a double shell containment of which the inner is made of prestressed and the outer of reinforced concrete,

while maintaining the passive function of decay heat removal. This subject is of interest also to other parties, and a number of European research centres are contributing to the programme.

The suggested decay heat removal system comprises a group of heat exchangers of special design installed in a particular fashion inside the primary containment and connected to another group of special heat exchangers located in a pool outside it. An air path in the space between the inner and outer containment shells will ensure the heat removal. An extensive test programme is underway at various ENEL laboratories and at Westinghouse STC.

2. The Athermip programme (Leaktightness tests of a full scale personnel airlock)

The objective of ensuring very low releases of radioactive material to the environment led to a careful analysis of all containment components, especially with respect to their leaktightness. In this context, the airlocks for passage of personnel and equipment are critical components. Therefore, such airlocks have been the subject of an ENEL/CISE activity which has received much international attention (by US DOE, EdF, European Union, etc.).

This activity was directed to study modifications that would make components apt to ensure leaktightness both at severe accident conditions and in long post accident transients. A test programme based on a real personnel passage airlock from the decommissioned Alto Lazio nuclear power plant has been prepared. For the tests, a metal shell that encloses the inner door, has been designed and provided with systems suitable for simulation of severe accident containment conditions. A thermal dissipation system has been installed around the outer door to cool it and to study the thermal and mechanical behaviour of the whole component as a function of the heat dissipation rate. In addition, the release rates through each of the seals will be measured.

3. Emergency injection system based on a "steam injector"

The feasibility study of a passive emergency injection system, based on a steam injector, is another research activity worth mentioning. A system has been studied, designed, tested and patented.

Development of calculation codes

Great attention has been dedicated to computational tools for reactivity accidents. Extensive studies have led to development, with the contribution of CISE, of the ENEL Integrated Code System (EICS). This system consists of a set of interconnected calculation programs, which enables ENEL to cope with problems associated with the design and operation of light water nuclear reactors with state-of-the-art computational tools. Another area for code development has been the invessel corium coolability, and a dedicated code called Corium 2-D is being validated after extensive development.

Tests for the AP-600 project

Two major test programmes for the development of the AP-600 project have been carried out in Italy with considerable assistance by ENEL.

1. Tests for the ADS (automatic depressurization system) function

The automatic depressurization of the primary circuit is among the most critical and innovative functions foreseen in AP-600 to face possible accidental scenarios. A series of tests have been carried out on the VAPORE circuit at ENEA's Casaccia CNR, simulating the operation of the ADS in a steam/water test circuit. These tests were performed to determine the dynamic loads on the structures of the Incontainment refuelling water storage tank (IRWST) induced by the discharge at prototype pressure and temperature conditions (ADS Step A), and also to confirm the valve and sparger operability at the discharge flow related to the AP-600 project (ADS Step B). Full scale valves and spargers were utilized in the tests.

2. Integrated high pressure tests of safety systems

The SPES-2 tests, carried out at SIET in the framework of a technical co-operation agreement between ENEL, ENEA, Ansaldo and Westinghouse, were planned to simulate the integrated thermo-hydraulic behaviour of the passive emergency systems and also their interaction with the active systems responsible for normal operation. The scale ratio against the real plant is 1:395 in volume and power, whereas pressure and vertical size are maintained, making it possible to accurately reproduce the actual pressure heads of paramount importance for the operation of the passive systems.

The experiments at SPES-2 included the simulation of such accidents as coolant leakage from the primary circuit, rupture of steam generator tubes, and the behaviour of the high pressure installation was especially examined. The data obtained have been utilized together with those from integral low pressure experiments which were carried out in a 1:4 scale of height at the Oregon State University, to validate the design codes, in particular those for licensing analyses to confirm the design certification of AP-600 apparatus.

Tests for the SBWR

In the framework of the international research and development programme on the SBWR reactor initiated by General Electric, and in addition to the ENEL/GE co-operation agreement for safety studies, an Ansaldo/ENEL/ENEA/GE co-operation agreement was established for carrying out tests on the prototypes of the isolation condenser (IC) and the condenser of the passive containment cooling system (PCCS). These two components represent the most innovative parts of the passive SBWR reactor.

The design of the two prototypes was made by Ansaldo; the first specifically made to enable the heat from the primary circuit to be removed during primary circuit isolating events, even when an ac and dc supply is absent, while the other shall remove heat from the containment atmosphere to the outside of the containment by water evaporation from a large pool. Taking into account the potential and flexibility of the test apparatus available there, SIET was selected to carry out the tests. The IC and PCCS circuits were simulated with special reference to the vertical dimensions of the apparatus reproducing actual elevation heads, making it possible to carry out the tests of the two prototypes under operational conditions in correspondence with the SBWR project.

3.2.2.5 ALWR licensing process

The Utility Requirements Document (URD) of the Electric Power Research Institute (EPRI) has been transmitted to and discussed with ANPA, the Italian safety authority. Recently, the revision B of the European Utility Requirements (EUR) has also been submitted to ANPA with the objective of having them discussed and to possibly obtaining an approval in co-ordination with other European Safety Authorities. Preparation of a Standard Safety Report for one or more plant designs of ENEL for submittal to ANPA is under consideration.

3.2.3 Netherlands

3.2.3.1 Current nuclear power plant capacity installed

In the Netherlands, 2 LWR plants, one BWR and one PWR plant, are currently in operation, with a total capacity of 504 MWe. The BWR plant at Dodeward, with a capacity of 55 MWe, was taken into operation in 1968, and the PWR plant at Borssele, with a capacity of 449 MWe, started operation in 1973, currently generate about 5% of the country's electricity. The percentage was higher in the past (when the electricity consumption was lower) and is gradually decreasing (with increasing demand).

3.2.3.2 Operating experience

The two Dutch nuclear power plants differ in type, capacity, objectives and prospects. On the other hand, they have several characteristics in common: they began operation quite early, have performed excellently with respect to availability, as can be seen from Figure 3.2-3, and recent probabilistic safety assessments (PSAs) indicate a high level of safety.

The Dodeward plant is a small BWR of which the objective, apart from generating electricity, was to stimulate the development of a national nuclear industry, to gain experience in the operation of a nuclear power plant, and to increase nuclear know-how. The design of this plant is somewhat special; it is the only nuclear power plant in operation worldwide that has invessel cooling by natural circulation. This passive safety feature has been adopted for the SBWR being developed by General Electric. Another special feature is the isolation condenser, which cools the core after an isolation scram, and which causes rapid depressurization of the reactor pressure vessel while maintaining the water level.

The Borssele PWR (KCB) near Vlissingen was taken into operation five years later than the Dodeward plant, and it has performed very reliably throughout the more than 20 years of operation with an average load factor of 80%.

A backfitting programme for the Dodeward BWR was initiated in the framework of the latest 10-yearly safety re-evaluation. The modification programme is focused primarily on increasing the redundancy of systems and components, on improving their physical separation, and on mitigation of the risk of severe accidents and external events. Input data for these studies consist of IAEA rules, operation experience, safety studies (including a Level 3 PSA) and of aging/surveillance experience. The backfitting programme aimed at completion in 1997; in October 1996, however, the utilities decided to shut down the Dodeward plant in March 1997.

Upgrading of the Borssele plant has been a more or less continuous process, and a complete Level 1, 2 and 3 probabilistic safety assessment performed for the plant indicates that the safety level of the plant, in terms of core melt frequency, is comparable to that of the latest generation of operating reactors. Following the practice currently developing in Western Europe to re-evaluate the plant safety every 10 years, the owner has initiated a modernization programme, by which KCB will be brought to compliance with recent nuclear regulations.

The defined hardware measures include: replacement of the primary safety/relief valves; improved functional capability of the ECCS and the cooling water trains; extension of the autonomous, bunkered cooling water system with groundwater pumps to provide an alternative ultimate heat sink; replacement of the main steam and feedwater lines from the steam generators up to the isolation valves to prevent large pipe ruptures; installation of larger diesel generators to improve the emergency power supply; extension of the reactor protection system and development of a special program for manmachine-interface aspects; and finally, provision of a filtered containment venting system for mitigation of postulated severe accidents.

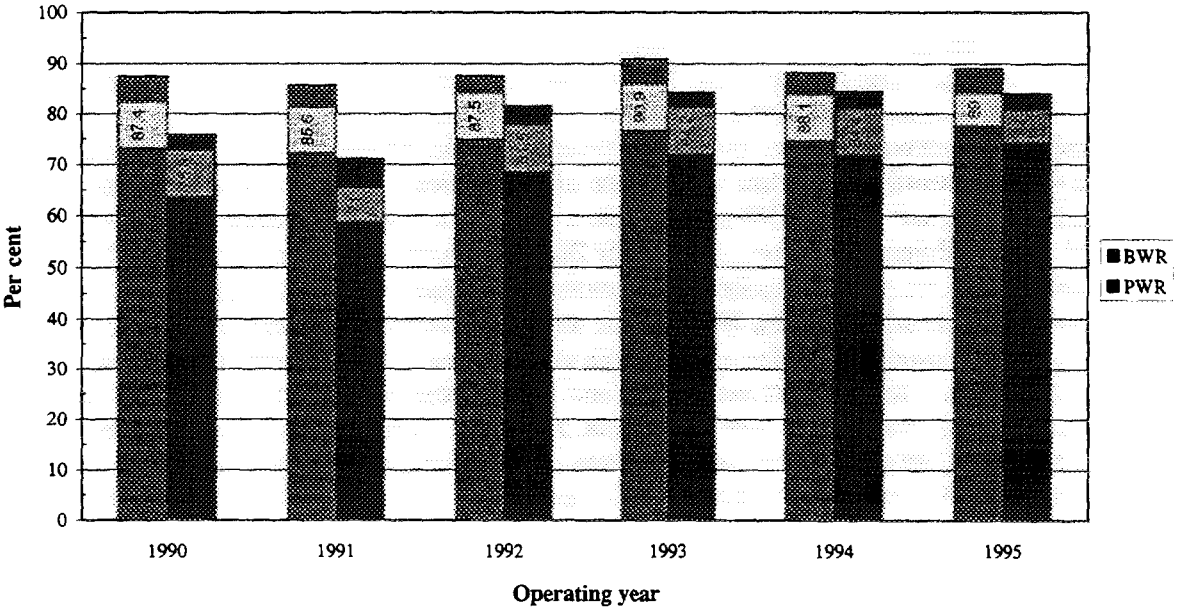


Figure 3.2.3 - Service time factors of Dutch nuclear power plants

The new safety concept will provide adequate protection for the variety of internal and external events generally postulated. The modification programme will enable KCB to continue operation until at least 2 004.

3.2.3.3 Nuclear power expansion plans

The Netherlands Electricity Act requires that an Electricity Plan shall be developed showing how the electricity demand is going to be met in the coming ten years period. In the most recent Electricity Plan (1995-2004) of the N.V. Sep/Dutch Electricity Generating Board the question of new nuclear power plants is not discussed at all due to the absence of any decisions by the central government on this issue. The Plan adheres to the policy of keeping the different options open, so that the coal fired and nuclear energy programmes can, if needed, be expanded in a responsible manner in the future. In view of the approach to keep the nuclear option open, the different "nuclear" parties in the Netherlands perform several tasks, such as:

- the Dutch utilities are extending the operating life of the two Dutch nuclear power plants;
- the safety authorities are working on a complete and consistent set of deterministic nuclear safety rules and regulations;
- the Dutch risk management policy regarding major accidents is elaborated for the application of individual and societal risk criteria on nuclear power plants;
- the Ministry of Economic Affairs subsidizes a programme that aims at supporting the Dutch utilities, industry and research organizations to keep up their nuclear competence and to participate in international efforts towards harmonization of requirements and evaluation of promising reactor designs; and
- the Dutch utilities join European activities in the field of common nuclear requirements (EUR) and familiarization with the design of advanced nuclear power plant designs for the European market.

The availability of natural gas has led to a substantial use of gas fired plants to generate electricity. In the utility sector, about 50% of the Dutch power output is coming from such plants, a further 40% is generated in coal fired plants, and 5% is produced by the nuclear power plants. The generating capacity

of the Dutch utilities does not suffice to meet the electricity demand of the country; almost 14% of the demand is currently covered by imports. Installation of additional generating capacity is not a high priority issue, however, since electricity supplies at competitive prices are available from other countries.

Despite the modest role of nuclear energy, the Dutch Government has always aimed to maintain nuclear competence, and therefore, the Ministry of Economic Affairs supports a "Programme to Enhance Nuclear Competence", PINK (an acronym for the Dutch programme title). The PINK programme is a programme in which five parties involved in nuclear energy in the Netherlands co-ordinate their efforts to intensify the nuclear competence of the utilities, research organizations and engineering companies. As a part of the PINK programme, a number of advanced reactor designs were studied in the mid 1980s with respect to their suitability for Dutch conditions and construction in the Netherlands, in some sort of an "options identification" programme as preparation for political discussions about the long term energy supply strategy in the Netherlands; after the Chernobyl accident in 1986, all nuclear plans were put aside, however. Further studies, with comparisons of evolutionary and innovative types, of added safety value, development perspective and the way the different reactor designs fulfill specific Dutch safety requirements have been carried out, however. These studies have confirmed that, at present, the modern, advanced water cooled reactors of evolutionary type represent the only realistic short term candidates for expansion of nuclear power generation in the Netherlands. As such, their further development and safety improvement should be the prime objective for Dutch nuclear development work.

Some viewpoints on the electricity production in the Netherlands, including a discussion of aspects like safety, waste, environmental effect, proliferation and societal issues, were presented in a document entitled "Dossier Kernenergie" that was published by the Dutch Government in December 1993. An important conclusion with respect to the risks and safety of future nuclear power plants was that in view of the current state of technology it is expected that most of the advanced reactor designs now under study will fulfill the proposed rules and regulations and will meet the Dutch risk criterion.

3.2.3.4 Development programme on advanced LWR

As noted above, there is no specific nuclear development programme in the Netherlands.

3.2.3.5 ALWR licensing process

The safety authorities are completing the Dutch nuclear codes and standards. The government, research institutes and the utilities are participating in international activities to harmonize codes, standards and requirements and make themselves acquainted with the design of reactors aimed for the European market.

3.2.4 Spain

3.2.4.1 Current nuclear power plant capacity installed

In Spain, a total of nine nuclear units are in operation, at seven sites, with a total capacity of 7400 MWe, representing 16,5% of the country's electricity generation capacity. These nine plants comprise seven PWR units, with a capacity of 5950 MWe, and two BWR units with a capacity of 1450 MWe. With respect to the NSSS design, six of the units are from Westinghouse, two from General Electric and one from Siemens KWU. In addition to the above, a GCR type plant belonging to the first generation was decommissioned in 1989.

The country's first three plants were negotiated as turnkey projects in the 1960s and began commercial operation at the end of that decade and the beginning of the 1970s. These plants constitute the so called first generation of plants in which construction participation by Spanish industry amounted to an average of around 40%. The second generation of plants includes five units which were ordered in the 1970s and entered into commercial operation during the 1980s. The participation by domestic industry was for these plants considerably higher, attaining figures of around 80%. Finally, there is a third generation comprising the two remaining plants, which was ordered at the end of the 1970s as multi-contract projects and with domestic participation in excess of 85%.

Five units with a joint output of 4 850 MWe were definitively canceled by a Spanish Government decision, after a moratorium period, while they were in different stages of completion, ranging from 5 to 99%.

The nuclear plants operate as base load units and have regularly contributed more than 35% of the country's electricity in recent years.

3.2.4.2 Operating experience

Considerable experience has been acquired in Spain through the construction of 15 nuclear power plants and the operation of 10; the national participation has attained high levels in most of these projects, as noted above. This places Spain in the position of a qualified nuclear country, with a high capacity for manufacturing, supply of services and research, and of an independent plant operator. Basically, the capacities acquired and developed cover the entire range of nuclear technologies, with the notable exception of the basic design of the NSSS.

Specifically, Spain possesses capacities in the fields of fuel research and supply, engineering and manufacturing, including know-how in relation to materials, lifetime management, simulation and optimization of burnup. With respect to uranium, there are abundant local reserves; the Spanish uranium company has also an 11% share in the Eurodif enrichment consortium. The country also possesses capacities in low and intermediate level radioactive waste management, with a laboratory integrated in the European network. Also available are capacities in the development of plant dismantling projects. Of special interest is the capacity to design storage and transport for spent fuel from the plants. In addition, the country provides very advanced services in relation to the inservice inspection and operating personnel training.

As regards nuclear safety, special mention may be made of the experience acquired in research and development in materials behaviour, safety related technology, probabilistic safety assessment and sever accidents management.

In relation to plant operating indicators, Spain has regularly occupied leading positions in the world availability ranking in recent years. Figure 3.2-4 shows the load and operating factors for the period 1990 to 1995, in global values per plant type.

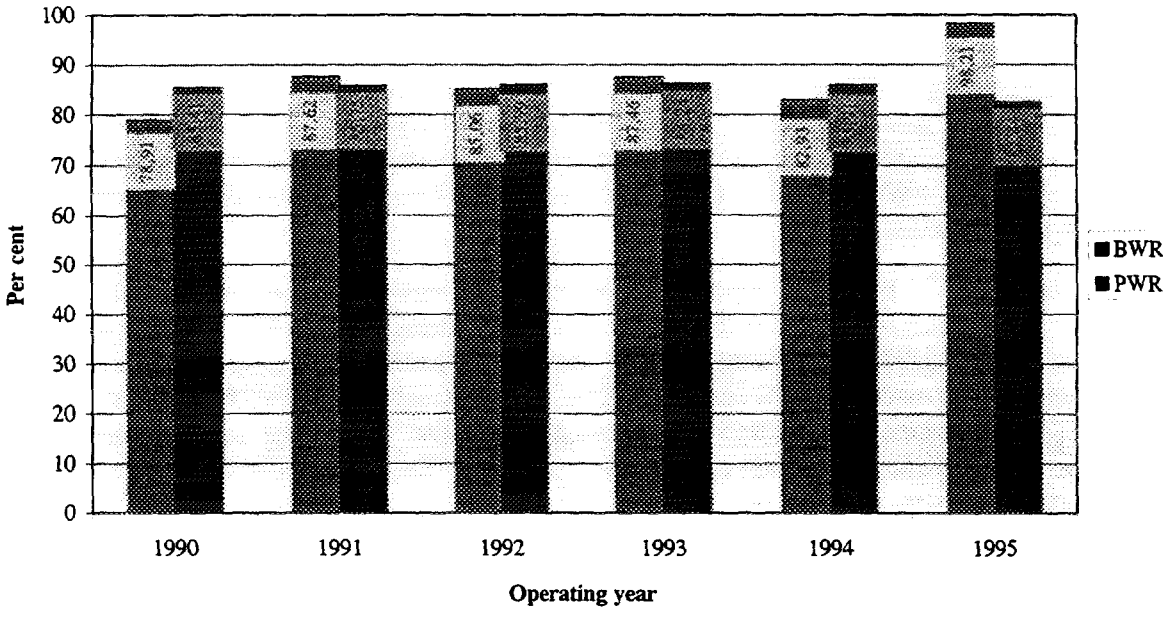


Figure 3.2-4 - Load factors of Spanish nuclear power plants

With the exception of 1995, the PWR plants present higher values than the BWRs. The following aspects are worthy of special mention; the PWR load and operating factors regularly show values of more than 85%, except for last year, when the steam generators were replaced in one of these plants. The two BWR plants on the other hand showed exceptional performance figures last year, with load and operating factors of close to 100%. The accumulated operating experience amounts to 160 reactor years.

3.2.4.3 Nuclear power expansion plans

As is the case in most Western countries, the Spanish nuclear programme is currently paralyzed for the same political, economic and social reasons that have led to the generalized nuclear moratorium in this part of the world. Nevertheless, nuclear energy is still considered to occupy an important place in electricity production in Spain, firstly due to optimized maintenance of the plants currently installed, and secondly, because it constitutes one of the most solid options from all points of view - safety, economy, and the environment - when there is a need to significantly increase installed power.

The total power installed in Spain as of December 1995 amounts to 44,64 GW. The average annual growth rate in the first half of the present decade has been approximately 30% lower than what was foreseen in the current National Energy Plan for 1991-2000, and a reduction in demand has been registered even in the industrial sector, reflecting the slowing down that has affected this type of activity. This situation means that there is an excess of installed capacity that will make expansion of the nuclear sector unnecessary, at least until the coming century.

In view of these circumstances, the nuclear strategy is oriented towards bringing about conditions that ensure the feasibility of this type of energy and its preparation for relaunching in the future. This is materialized by way of the following specific objectives: (1) to prepare future solutions through the participation in international advanced nuclear projects, and (2) to contribute to the maintenance of nuclear technology and of the capacities acquired to date, both with a view of undertaking future nuclear projects and of providing the support required for the plants currently in operation.

In conclusion, it may be stated that although there are currently no plans for expansion in Spain, neither nuclear nor other, the performance of certain research and development activities in the nuclear area is considered a necessity and even a priority issue, in order to be prepared for an imminent

relaunching of nuclear energy that could happen around the turn of the century. In planning these activities, consideration should be given to the most significant international trends and to the aspects differentiating the Spanish situation, characterized by the lack of a national NSSS designer, the operation of nuclear power plants of mainly US design and the country's geopolitical conditions as part of the European Union.

3.2.4.4 Development programme on advanced LWR

This programme is conceived as constituting co-ordinated action by the electricity sector and the nuclear industry, structured along two main lines depending on whether orientation is towards American technology or the European environment. Basically, the programme is made up of three projects:

- The light water passive plant project consists of participation in design, design certification and detailed engineering activities relating to the General Electric and Westinghouse plants under development within the EPRI Advanced LWR program. The project is being developed in two phases: basic design and detailed engineering design. It includes some one hundred activities distributed in the technological areas of fuel, the primary system, engineering, man-machine interface, inservice inspection, containment and safety assessment.

- The European Utilities Requirements (EUR) project consists of a contribution to the preparation of the requirements document of the European utilities in relation to the nuclear power plants to be built on this continent in the future. This contribution includes overall review of the document, and authorship of four chapters: 2.2, Performance Requirements; 2.5, Codes and Standards; 2.6, Material Related Requirements; and 2.12, Operation, Maintenance and Procedures.

- The European Passive Reactor project consists of participation, along with other European electricity utilities, in the conceptual design of a possible adaptation of the AP-600 reactor to EUR requirements, including upgrading to the 1000 MW range.

Spain is also considering collaboration in other projects involving European technologies, within a framework of participation at the country's level of technology and with clear objectiveness of competitiveness and safety.

3.2.4.5 ALWR licensing process

The licensing process currently applied in Spain uses the US 10CFR50 regulation as the main reference. The most relevant aspects of this regulation may be summarized as follows: there are two stages in the licensing, namely the construction permit and the operating license that is awarded after the plant has been built; and a complete and individualized licensing process is required for each plant.

For the time being, no changes are anticipated to the regulatory framework that might allow a process different from that described above to be applied to advanced plants. In addition, as there are no immediate plans to build new plants in Spain, there is no urgency requirement for the regulatory processes to be revised. Nevertheless, the evolution of this issue in both the United States and Europe is followed closely.

3.2.5 Switzerland

3.2.5.1 Current nuclear power plant capacity installed

In Switzerland, five LWR plants are currently in operation, with a total capacity of 3050 MWe. These plants comprise two BWR and three PWR units with a capacity of 1385 and 1665 MWe, respectively.

The first two Swiss nuclear power plants, Beznau and Mühleberg, each belongs to a single large public electric utility, whereas the two later plants, Gösgen and Leibstadt, are partner plants of several electric utilities and public service companies. The concept of partner nuclear power plants made it possible, when they were set up in the seventies, for medium sized and smaller organizations to share in economically attractive, large scale power generation plants and to gain access to the latest technology. In each case, one of the partners has the responsibility for the business management on behalf of the others.

Some statistical data for the Swiss nuclear power plants, related to the 1995 operating season, are shown in Table 3.2.5-I.

3.2.5.2 Operating experience

In 1995, the Swiss nuclear power plants produced their highest ever combined annual output, 23,57 TWh. Their contribution to total electricity generation in the country was 38.9%. The average service time factors for the Swiss nuclear power plants during the last six years, that are shown in Figure 3.2-5 on next page, illustrate the good operating record of the plants.

The nominal net capacity of the Gösgen and Leibstadt plants was increased by 25 and 40 MW, respectively, on January 1, 1995. In both Beznau units, the high pressure turbines were replaced by turbines with higher efficiency. After replacement of the third low pressure turbine, the nominal net capacity of Gösgen was increased by a further 5 MW. The new value is set at 970 MW from January 1996.

After an extended operating cycle of 17 months - for the first time in Switzerland - at unit 2 of the Beznau Nuclear Power Station, the refuelling outage in this unit lasted 61 days. The refuelling shut-downs in the other four Swiss units were all short, lasting 35 days or less; at Gösgen, the refuelling outage was only 27 days.

The average number of reactor scrams at the Swiss plants remained stable, at less than one scram per reactor year.

Extensive testing of the core shroud of the Mühleberg plant revealed measurable crack growth at two previously detected cracks. Although the safety of the plant is not affected in any way, preventive measures were decided upon to ensure the structural stability of the core shroud.

Table 3.2.5-I Swiss nuclear power plant statistics

Power plant	Type	Net power MWe	Commis-sioning	Production in 1995 [TWh]	Availability in 1995 [%]
Beznau I	PWR	350	1969	2.82	92.6
Beznau II	PWR	350	1971	2.55	83.3
Mühleberg	BWR	355	1971	2.67	85.8
Gösgen	PWR	965	1979	7.77	92.5
Leibstadt	BWR	1030	1984	7.68	85.1
		3050		23.49	88.2 [weighted] 87.9 [arithmetic]

The Beznau and Gösgen plants have also produced process and district heat (1000 TJ in 1995).

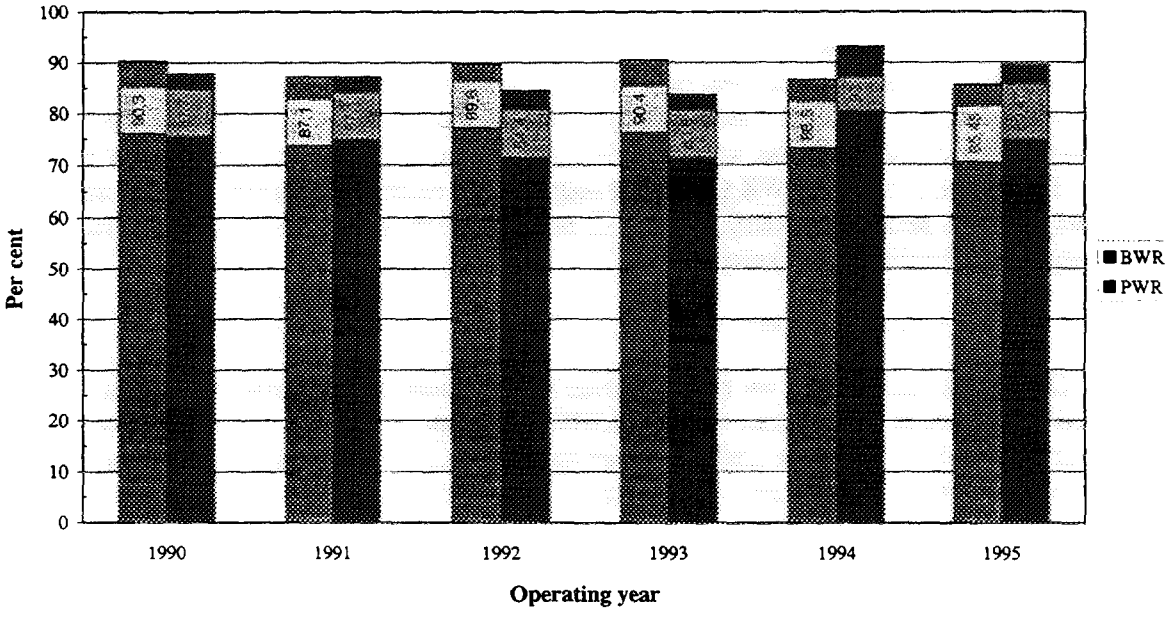


Figure 3.2-5 - Service time factors of Swiss nuclear power plants

At Beznau II, the decision was made to replace the steam generators, too. The application for a power uprate of the Leibstadt reactor is still pending.

An IAEA Operational Safety Review Team (OSART) commended the excellent operating history of the two unit Beznau nuclear power station, its condition and the qualification of the personnel at all levels.

3.2.5.3 Nuclear power expansion plans

Due to the political situation, there are no nuclear power expansion plans. The nuclear utilities strategy is to safely operate the existing plants until the end of their technical lifespan and to maintain the option to replace the present nuclear power plants by new plants of the same capacity and at the same sites.

3.2.5.4 Development programme on advanced LWR

Nuclear fission R&D concentrates on safety related topics of today's light water reactors, on passive safety issues of advanced reactor concepts (within international co-operation programmes) and on the disposal of radioactive wastes. The federal budget for nuclear fission R&D was about 30 million Swiss francs in 1995. It is assumed to be further reduced in real terms in the coming years. The major portion of the research is carried out at the Paul Scherrer Institute (PSI). Some work is also performed at the Swiss Federal Institutes of Technology in Zürich and Lausanne and at the industry.

The LWR safety research programme is centred on source term analysis of Swiss nuclear power plants, transient analysis of Swiss nuclear power plants, and on nuclear power plant life extension (material problems). Further effort is invested in safety related operational issues of existing nuclear power plants, e.g., primary water contamination, PIE. Research activities are conducted in the framework of international co-operation.

The waste management research programme mainly focuses on performance and safety assessment of waste repositories, i.e., characterization of waste forms, and repository near field and far field studies. Emphasis is put on development of models of relevant mechanisms for nuclide transport in the

geosphere and their validation by experiments, and on data acquisition for safety analysis. The work is done in close co-operation with NAGRA.

The research programme on the safety of future reactor systems concentrates mainly on topics of advanced light water reactors (LWRs). A large experimental facility, PANDA, has been constructed for the investigation of passive decay heat removal and fission product retention in advanced LWRs, in close co-operation with the US EPRI research programme and with the financial support of the Swiss utilities. The 5-year project, named Alpha, started in 1992, and the facility started its operation at the beginning of 1995. Experiments for the SBWR (simplified boiling water reactor) of General Electric were conducted in 1995 and will be analyzed during 1996. A programme for a next step, using the facility for new European reactor concepts, is in preparation.

3.2.5.5 ALWR licensing process

Due to the moratorium on nuclear power development, no new reactors are at present being planned or licensed in Switzerland. If a new LWR plant would go through licensing, however, the HSK-101 design guidelines would be applied. When assessing existing installations, these guidelines are also consulted to identify discrepancies. Observance of the guidelines facilitates the evidence of safety, while discrepancies must be judged and assessed on a case-by-case basis.

3.2.6 The Czech Republic

3.2.6.1 Current nuclear power plant capacity installed

In the Czech Republic, four nuclear power plant units are in operation at the site of Dukovany in the south-east part of the country (the region of Moravia). The units are of the Russian WWER-440, version V 213, design, with a total installed capacity of 1760 MWe. They were built in the early 1980s and taken into operation in the period 1985-1987; the main components were manufactured by the Czech and Slovak industry.

3.2.6.2 Operating experience

All units of the Dukovany nuclear power station have shown successful and reliable operation, as can be seen by the average load factors recorded during the last six years (Figure 3.2-6). The first unit passed "10 years of operation" in 1995, and the average load factor for the station, over these 10 years, amounts to above 79%. In accordance with a decision of the Czech State Office on Nuclear Safety (SONS), the nuclear safety of the first unit was re-evaluated after the 10 years of operation and an updated Safety Report was elaborated. On the basis of this report, a license for continued operation of the unit has been issued by SONS.

The same procedure is envisaged for the three other units which are approaching their "10 years of operation" dates.

A programme of upgrading the Dukovany nuclear power station was initiated in 1983, aiming at extension of the lifetime of the units and at enhanced nuclear safety.

3.2.6.3 Nuclear power expansion plans

Two WWER-1000 units have been under construction at Temelin since 1987. In line with recommendations of international missions, several improvements to the design of the units have been introduced; the construction work is now being realized on a contractual basis in co-operation with the Westinghouse Electric Corporation.

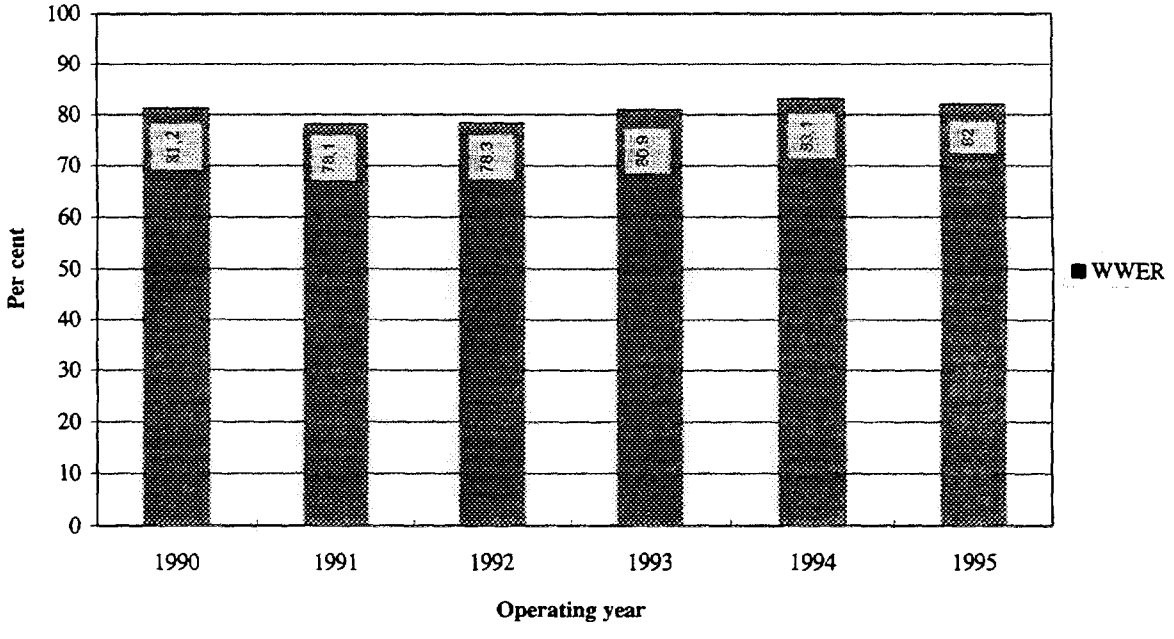


Figure 3.2-6 - Load factors of nuclear power plants in the Czech Republic

The design modernization comprises:

- fuel;
- I&C system;
- radiation monitoring system; and
- technical diagnostic system.

An updated licensing process, based on current world practice, has been applied for the safety review of the Temelin units.

There are at present no plans for further nuclear power plant construction in the country.

3.2.6.4 Development programme on advanced LWR

The Czech industry completed in 1990 a feasibility study of an as advanced 600 MWe unit based on a combination of the WWER and advanced technology from Western countries, but after that, all Czech Republic activities in the field of really advanced nuclear projects have been interrupted. Czech industry and research organizations participate in some partial research tasks, however.

Principal conditions that would lead to revival of considerations on new nuclear power plants, unquestionably of advanced design, are:

- Successful commissioning and operation of the Temelin nuclear power station; and
- successful construction and operation of an advanced nuclear unit in another country, most preferably in a country with a leading role in nuclear technology.

3.2.6.5 ALWR licensing process

The Czech Republic has adopted an up graded licensing process that is based on current world practice, as noted in Section 3.2.6.3.

CHAPTER 4. EVOLUTIONARY, LARGE SIZE ALWR DESIGNS (850 MWe or larger)

4.1 ABWR, GE, USA IN CO-OPERATION WITH HITACHI AND TOSHIBA, JAPAN

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

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

The following design description which has been provided for this report by General Electric, reflects the US version of the ABWR design which differs slightly from the Japanese version. The US ABWR version received, as mentioned in Section 1.3.2, a design certification from the US NRC in May 1997.

4.1.1 Introduction

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

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

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

General power generation design criteria

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

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

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

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

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

General safety design criteria

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

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

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

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

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

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

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

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

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

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

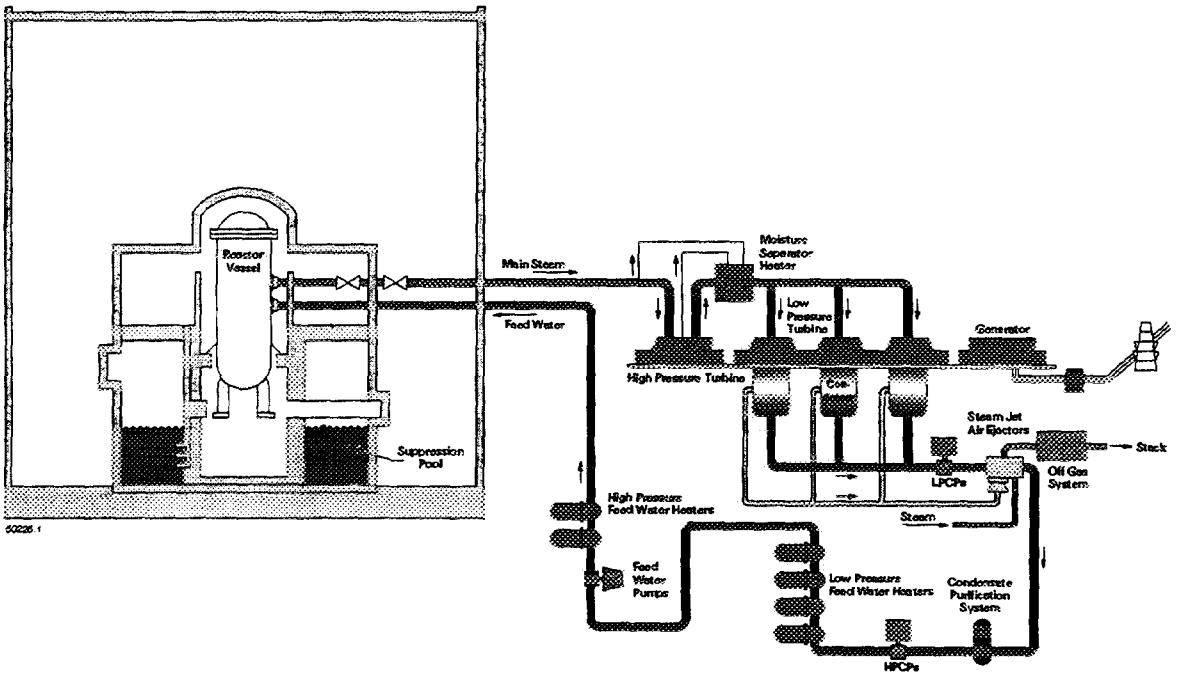


Figure 4.1-1 ABWR - Steam cycle

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

4.1.2 Description of the nuclear systems

4.1.2.1 Primary circuit and its main characteristics

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

(1) to deliver steam from the reactor pressure vessel (RPV) to the turbine main steam system,

(2) to deliver feedwater from the condensate and feedwater system to the RPV,

(3) to provide overpressure protection of the reactor coolant pressure boundary,

(4) to provide automatic depressurization of the RPV in the event of a loss of coolant accident (LOCA) where the RPV does not depressurize rapidly, and

(5) with the exception of monitoring the neutron flux, to provide the instrumentation necessary for monitoring conditions in the RPV such as RPV pressure, metal temperature, and water level instrumentation.

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

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

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

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

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

4.1.2.2 Reactor core and fuel design

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

Control rod drive system

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

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

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

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

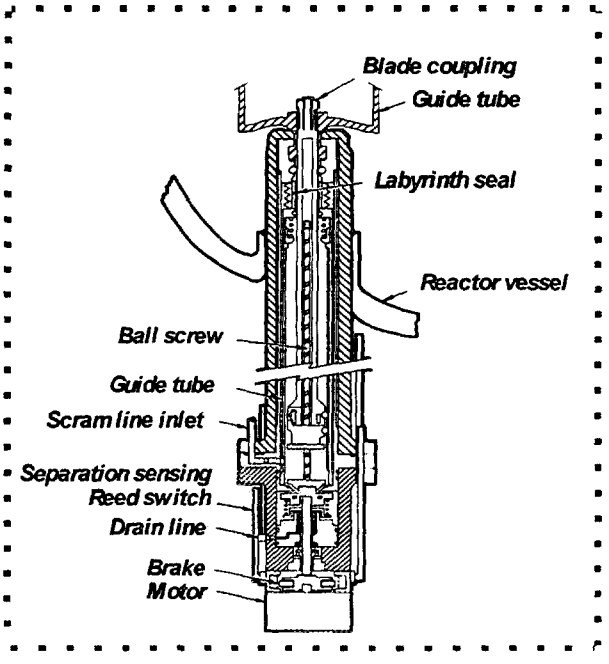


Figure 4.1-2 Cross-section of fine motion control rod drive

4.1.2.3 Fuel handling and transfer systems

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

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

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

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

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

4.1.2.4 Primary components

Reactor pressure vessel

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

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

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

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

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

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

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

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

Reactor internals

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

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

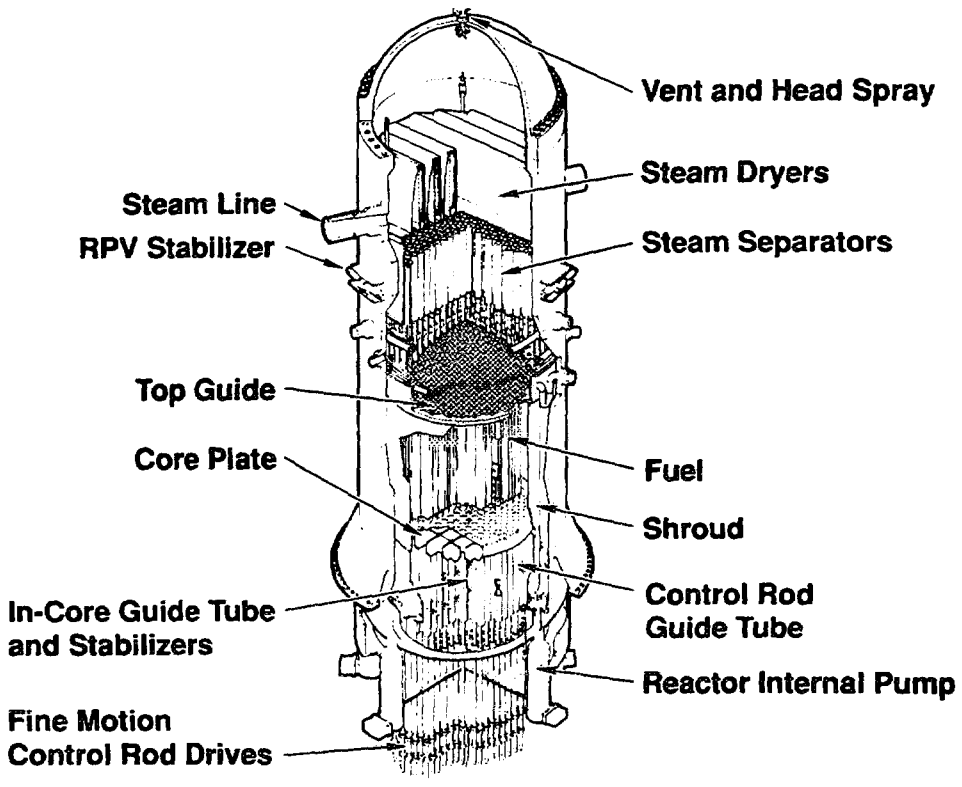


Figure 4.1-3 ABWR - Reactor pressure vessel and internals

Other reactor internals are:

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

Reactor recirculation pumps

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

The recirculation flow rate is variable over a "flow control range," from minimum flow established by certain pump performance characteristics to above maximum flow required to obtain rated reactor power. By regulating the flow rate, the reactor power output can be regulated over an approximate range from 70 to 100% of rated output, without moving control rods. RIP performance is adequate to allow plant operation at 100% power with only 9 of the 10 pumps in operation.

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

4.1.2.5 Reactor auxiliary systems

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

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

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

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

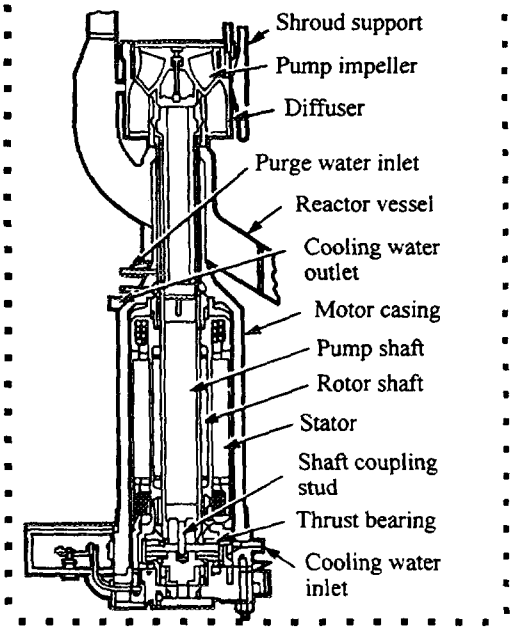


Figure 4.1-4 Cross-section of Reactor internal pump

shared by the SPCU system for cleaning the suppression pool water. The FPCU and SPCU systems are basically the same as on previous BWRs.

4.1.2.6 Operating characteristics

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

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

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

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

4.1.3 Description of turbine generator plant system

4.1.3.1 Turbine generator plant

The main turbine

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

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

Turbine bypass system

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

The TBP consists of a three-valve chest that is connected to the main steam lines upstream of the turbine stop valves, and of three dump lines that separately connect each bypass valve outlet to one

condenser shell. The system is designed to bypass at least 33% of the rated main steam flow directly to the condenser. The TBP, in combination with the reactor systems, provides the capability to shed 40% of the turbine generator rated load without reactor trip and without the operation of safety/relief valves. A load rejection in excess of 40% is expected to result in reactor trip but without operation of any steam safety valve.

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

Main condenser

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

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

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

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

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

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

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

abnormal radioactivity is detected. Radiation monitors are provided on the main steamlines which trip the vacuum pump if abnormal radioactivity is detected in the steam being supplied to the condenser.

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

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

4.1.3.2 Condensate and feedwater systems

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

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

4.1.3.3 Auxiliary systems

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

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

4.1.4 Instrumentation and control systems

4.1.4.1 Design concepts including control room

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

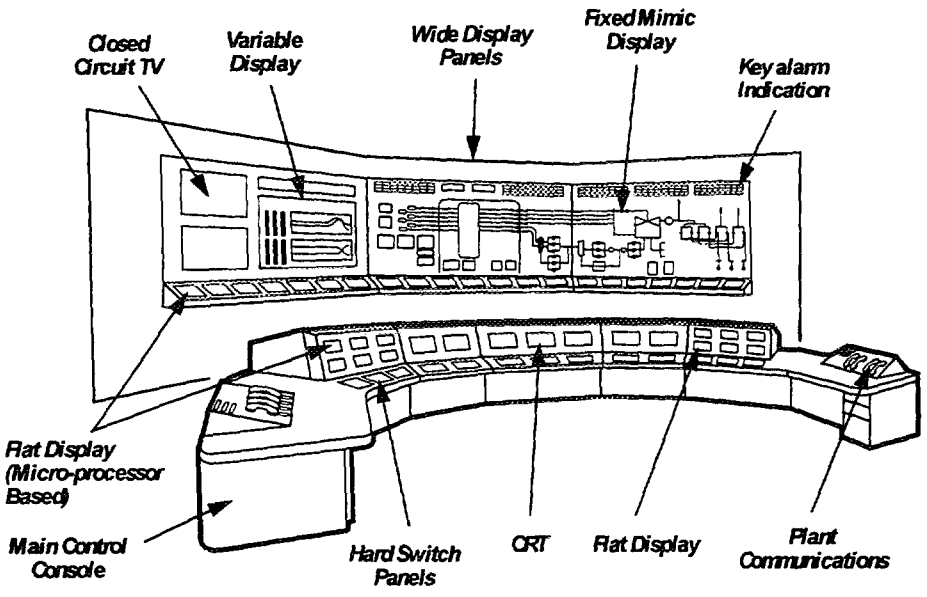


Figure 4.1-5 ABWR - Main control room features

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

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

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

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

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

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

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

4.1.4.2 Reactor protection system and other safety systems

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

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

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

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

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

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

Startup range neutron monitoring (SRNM) subsystem

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

Power range neutron monitoring (PRNM) subsystem

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

The PRNM subsystem consists of two subsystems:

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

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

Automatic traversing in-core probe (ATIP) subsystem

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

Multi-channel rod block monitor (MRBM) subsystem

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

4.1.5 Electrical systems

4.1.5.1 Operational power supply systems

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

Direct current power supply

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

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

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

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

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

Instrument and control power supply

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

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

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

4.1.5.2 Safety-related systems

Class IE AC power supply

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

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

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

Direct current power supply

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

Vital (uninterruptable) power supply

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

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

4.1.6 Safety concept

4.1.6.1 Safety requirements and design philosophy

Recognizing the need for continued safety enhancements in plant operation, one of GE's goals in designing the ABWR was to reduce core damage frequency by at least an order of magnitude relative to currently operating GE plants. The most important design feature contributing to this was the adoption of reactor internal pumps (RIPs). These vessel-mounted pumps eliminate large attached recirculation piping, particularly involving penetrations below the top of the core elevation, and make it possible for a smaller Emergency core cooling system (ECCS) network to maintain core coverage during a postulated loss of coolant accident.

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

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

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

4.1.6.3 Severe accidents (beyond design basis accidents)

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

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

4.1.7 Plant layout

4.1.7.1 Buildings and structures, including plot plan

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

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

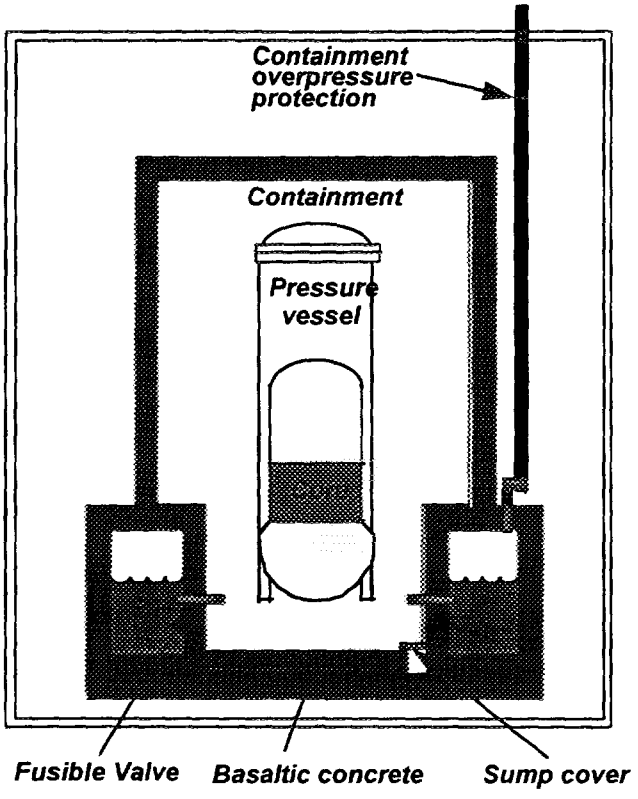


Figure 4.1-6 ABWR - Severe accident mitigation features

(b) Service building - personnel facilities, security offices, and health physics station.

(c) Control building - includes the control room, the computer facility, reactor building component cooling water system and the control room HVAC system.

(d) Turbine building - houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.

(e) Radwaste building - houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

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

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

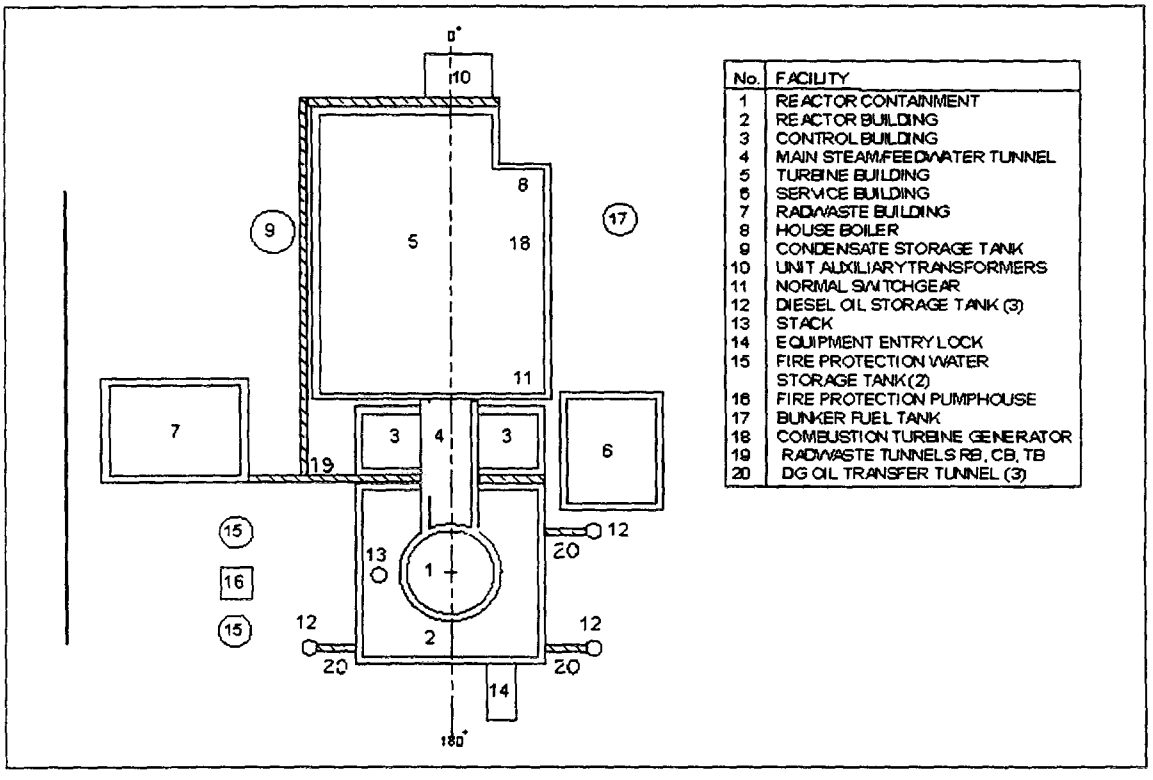
(a) Retain the passive and well established BWR pressure suppression containment technology. Use of the horizontal vent configuration confirmed for the Mark III containments.

(b) Emphasize optimal layout of systems to improve personnel access and equipment maintenance activities.

(c) Locate major equipment for early installation using open top construction approach and large scale modularization.

(d) Arrange the reactor building around the primary containment to provide multiple barriers to post-accident fission product leakage, and high tolerance to external missiles.

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



No.	FACILITY
1	REACTOR CONTAINMENT
2	REACTOR BUILDING
3	CONTROL BUILDING
4	MAIN STEAM/FEEDWATER TUNNEL
5	TURBINE BUILDING
6	SERVICE BUILDING
7	RADWASTE BUILDING
8	HOUSE BOILER
9	CONDENSATE STORAGE TANK
10	UNIT AUXILIARY TRANSFORMERS
11	NORMAL SWITCHGEAR
12	DIESEL OIL STORAGE TANK (3)
13	STACK
14	EQUIPMENT ENTRY LOCK
15	FIRE PROTECTION WATER STORAGE TANK (2)
16	FIRE PROTECTION PUMPHOUSE
17	BUNKER FUEL TANK
18	COMBUSTION TURBINE GENERATOR
19	RADWASTE TUNNELS RB, CB, TB
20	DG OIL TRANSFER TUNNEL (3)

Figure 4.1-7 ABWR site plan

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

(a) Personnel access for all normal operating and maintenance activities was a primary concern starting with the first layout studies. Access routes from the change room to contaminated reactor and turbine building areas are as direct as possible. At each floor, 360° access is provided, if practical, to enhance daily inspections and normal work activities. Access to equipment not reachable from floor level is via platform and stair access wherever possible.

(b) Equipment access is provided for all surveillance, maintenance and replacement activities with local service areas and laydown space for periodic inspections. Adequate hallways and other equipment removal paths, including vertical access hatches, are provided for moving equipment from its installed position to service areas or out of the building for repair. Lifting points, monorails and other installed devices are provided to facilitate equipment handling and minimize the need for re-rigging individual equipment movements. The equipment access also considers the need for construction access.

(c) Radiation levels for personnel are controlled and minimized. The reactor building is divided into clean and controlled areas. Once personnel enter a clean or controlled area, it is not possible to crossover to the other area without returning to the change area. Redundant equipment is located in shielded cells to permit servicing one piece of equipment while the plant continues to operate. Valve galleries are provided to minimize personnel exposure during system operation or preparation for maintenance.

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

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

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

4.1.7.2 Reactor building

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

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

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

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

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

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

Key distinguishing features of the ABWR reactor building design include:

(a) Elimination of external recirculation loops reduces the containment volume associated with high construction costs.

(b) Reduced building volume reduces material costs and construction schedule.

(c) Designed with simple structural shapes to improve constructability to reduce capital costs and the construction schedule.

(d) Improved personnel and equipment access for enhanced operability and maintainability.

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

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

4.1.7.3 Containment

The ABWR pressure suppression primary containment system, which comprises the drywell (DW), wetwell, and supporting systems, is designed to have the following functional capabilities:

(a) The containment structure is designed to maintain its functional integrity during and following the peak transient pressures and temperatures which would occur following any postulated loss of coolant accident (LOCA). A design basis accident (DBA) is defined as the worst LOCA pipe break (which leads to maximum containment (and DW) pressure and/or temperature), and is further postulated to occur simultaneously with a loss of off site power (LOOP) and a safe shutdown earthquake (SSE).

The containment structure is designed for the full range of loading conditions consistent with normal plant operating and accident conditions including the LOCA related design loads in and above the suppression pool (SP).

The containment structure is designed to accommodate the negative pressure difference between the drywell and wetwell and relative to the reactor building surrounding.

(b) The containment structure and isolation, with concurrent operation of other accident mitigation systems, is designed to limit fission product leakage during and following the postulated design basis accident (DBA) to values less than leakage rates which would result in off-site radiation doses greater than those set forth in 10CFR100.

(c) Capability for rapid closure or isolation of all pipes or ducts which penetrate the containment boundary is provided to maintain leakage within acceptable limits.

(d) The containment structure is designed to withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.

(e) The containment structure is designed to accommodate flooding to a sufficient depth above the active fuel to permit safe removal of the fuel assemblies from the reactor core after the postulated DBA.

(f) The containment structure is protected from or designed to withstand hypothetical missiles from internal sources and uncontrolled motion of broken pipes which could endanger the integrity of the containment.

(g) The containment structure is designed to provide means to channel the flow from postulated pipe ruptures in the drywell to the suppression pool.

(h) The containment system is designed to allow for periodic tests at the calculated peak or reduced test pressure to measure the leakage from individual penetrations and isolation valves, and the integrated leakage rate from the structure to confirm the leaktight integrity of the containment.

(i) The atmospheric control system (ACS) establishes and maintains the containment atmosphere to less than 3.5% (by volume) oxygen during normal operating conditions. To assure an inert atmosphere, operation of two permanently installed recombiners can be initiated on high levels as determined by the containment atmospheric monitoring system (CAMS).

The main features of the ABWR containment structure are illustrated in Figure 4.1-8.

4.1.7.4 Turbine building

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

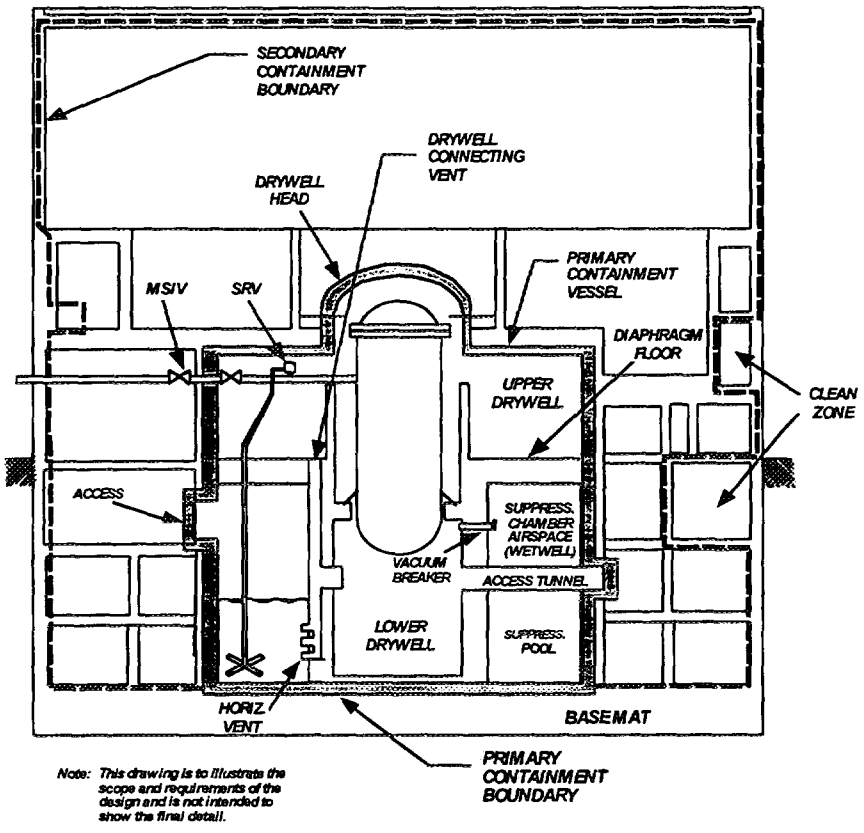


Figure 4.1-8. ABWR - Containment structure features

4.1.7.5 Other buildings

No information provided.

4.1.8 Technical data (for the US Version of ABWR)

General plant data		
Power plant output, gross	1385	MWe
Power plant output, net	1300	MWe
Reactor thermal output	3926	MWth
Power plant efficiency, net	33.1	%
Cooling water temperature	≈ 28.0	°C
Nuclear steam supply system		
Number of coolant loops	1	
Primary circuit volume		m^3
Steam flow rate at nominal conditions	2122	kg/s
Feedwater flow rate at nominal conditions	2118	kg/s
Reactor coolant system		
Primary coolant flow rate	14502	kg/s
Reactor operating pressure	7.17	MPa
Steam temperature/pressure	287.8/7.17	°C/MPa
Feedwater temperature	215.6	°C
Core coolant inlet temperature	278	°C
Core coolant outlet temperature	288	°C
Mean temperature rise across core	10	°C
Reactor core (GEP 8x8R Fuel Design)		
Active core height	3.710	m
Equivalent core diameter	5.164	m
Heat transfer surface in the core		m^2
Average linear heat rate	234.7	kW/m
Fuel weight	172	t U
Average fuel power density		kW/kg U
Average core power density	50.6	kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		
Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	4470	mm
Rod array	8x8, square lattice	
Number of fuel assemblies	872	
Number of fuel rods/assembly	62	
Number of spacers		
Enrichment (range) of first core, average	2.0 (appr.)	Wt%
Enrichment of reload fuel at equilibrium core	3 to 4	Wt%
Operating cycle length (fuel cycle length)	24	months
Average discharge burnup of fuel [capability]	>50,000	MWd/*t*
Cladding tube material	annealed, recrystallised Zr 2	
Cladding tube wall thickness		
Outer diameter of fuel rods	12.3	mm
Fuel channel/box; material	Zr-4	
Overall weight of assembly, including box	306	kg
Uranium weight/assembly	197	kg
Active length of fuel rods	3.710	mm
Burnable absorber, strategy/material	axial and radial grading/ Gd_2O_3 mixed with fuel	
Number of control rods	205	
Absorber material	B_4C and Hafnium	
Drive mechanism	electro-mechanical/hydraulic	
Positioning rate	30	mm/*s*
Soluble neutron absorber	Boron	
Reactor pressure vessel		
Inner diameter of cylindrical shell	7 100	mm
Wall thickness of cylindrical shell	190	mm
Total height, inside	21 000	mm
Base material: cylindrical shell	low-alloy carbon steel [to ASTM A533, grade B, ASTM A508, class 3, or equiv.]	
RPV head	stainless steel	
lining		
Design pressure/temperature	8.62/301.7	MPa/°C
Transport weight (lower part w/rigging)	1164	t
RPV head	≈100	t

Reactor recirculation pump

Type	variable speed, wet motor, single stage, vertical internal pump	
Number	10	
Design pressure/temperature	same as for RPV	MPa/°C
Design mass flow rate (at operating conditions)	1624 (each)	kg/s
Pump head	0.287	MPa
Rated power of pump motor (nominal flow rate)	≈800	kW
Pump casing material	same as for RPV	
Pump speed (at rated conditions)	≤1500	rpm
Pump inertia		kg m^2

Primary containment

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

Reactor auxiliary systems

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

Power supply systems

Main transformer,	rated voltage	24/525	kV
	rated capacity	1660	MVA
Plant transformers,	rated voltage	24/4.16/13.8	kV
	rated capacity	50/15/35	MVA
Start-up transformer	rated voltage		kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)		13.8/4.16	kV
Number of low voltage busbar systems			
Standby diesel generating units: number		3	
rated power		6.57	MW
Number of diesel-backed busbar systems		3	
Voltage level of these		4160	V AC
Number of DC distributions			
Voltage level of these			V DC
Number of battery-backed busbar systems			
Voltage level of these			V AC

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	six flow, tandem compound, single reheat	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/3 LP	
Turbine speed	1800	rpm
Overall length of turbine unit		m
Overall width of turbine unit	m	
HP inlet pressure/temperature	6.792/283.7	MPa/°C

Generator

Type	3-phase, turbo-generator	
Rated power	1620	MVA
Active power	1385	MW
Voltage	24	24 kV
Frequency	60	Hz
Total generator mass, including exciter		t
Overall length of generator	m	

Condenser

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

Condensate pumps		
Number	4 x 50%	
Flow rate	≈ 435	kg/s
Pump head	3.82	MPa
Temperature		°C
Pump speed		rpm
Condensate clean-up system		
Full flow/part flow	Full flow	
Filter type (deep bed or rod type)	deep bed	
Feedwater tank		
Volume		m^3
Pressure/temperature		MPa/°C
Feedwater pumps		
Number	3 × 65%	
Flow rate	≈ 1000	kg/s
Pump head	6	MPa
Feed pump power		MW
Feedwater temperature (final)	216	°C
Pump speed		rpm
Condensate and feedwater heaters		
Number of heating stages, low pressure	3 × 4	
high pressure	2 × 2	
feedwater tank		

4.1.9 Project status and planned schedule

4.1.9.1 Entities involved (Japan)

The Japanese version of the ABWR was developed by GE and its associates Hitachi Ltd. and Toshiba Corp. under the sponsorship of the Tokyo Electric Power Company (TEPCO). In 1987, TEPCO announced its decision to proceed with a two-unit ABWR project at its Kashiwazaki-Kariwa nuclear power station.

4.1.9.2 Entities involved (US)

First of a kind engineering (FOAKE) is developing the ABWR design for application in the United States. Funding for the ABWR FOAKE project is provided by GE and its FOAKE associates including members of the ABWR FOAKE design team, the advanced reactor Corporation (ARC), representing utility sponsors of the ABWR FOAKE project, and the United States Department of Energy (DOE). The ABWR FOAKE Project began in June 1993 and is on schedule for completion in September 1996[1].

The ABWR FOAKE participating utilities include Commonwealth Edison Company, Carolina Power and Light Company, General Public Utilities (GPU), Tennessee Valley Authority (TVA), TU Electric Company, Florida Power and Light Company, Duke Power Company, Public Service Electric and Gas (PSEG), Electricite de France (EdF), and Agrupacion Electrica para el Desarrollo Tecnologico Nuclear - A.I.E. (DTN - representing Spanish utilities). Personnel from the Institute of Nuclear Power Operations (INPO) and EPRI joined the ARC and utility personnel in providing lessons learned input from construction, startup and operations to the project through the review of deliverables, design reviews, and topical technical meetings.

Early in the ABWR FOAKE effort a decision was made to carry out the work with a group of companies (the Design Team) who could not only do the ABWR FOAKE design work but who could potentially work together, or with others, in future ventures to deliver ABWR construction projects worldwide.

The ABWR FOAKE design team consists of GE, Bechtel Power Corporation (Bechtel), Black and Veatch, Adtechs Corporation - a subsidiary of Japan Gas Company (JGC), Chicago Bridge and Iron Technical Services Company (CB&I), Shimizu Corporation, Hitachi Ltd., Toshiba Corporation, Badan Tenaga Atom Nasional (BATAN- the national atomic energy agency of Indonesia), Comision Federal de Electricidad (CFE - the national electric power utility of Mexico), Simulation Systems and Services Technologies Company (S3 Technologies), Equipos Nucleares S.A. (ENSA), Tecnatom, S.A., and UTE-INITEC/Empresarios Agrupados (UTE).

- GE has overall responsibility for the project and is responsible for the detailed design of the reactor and control buildings, including all of the systems and equipment in those buildings. GE is also responsible for the cost estimate and construction schedule with support from all other participants in their assigned areas of responsibility. GE Power Generation is responsible for the turbine and generator design.
- Bechtel provides support to GE in the areas of mechanical and control and electrical systems design and civil design in the reactor and control buildings. Bechtel is also responsible for the service building design.
- Black and Veatch is responsible for the turbine island design. Black and Veatch also provides the POWRTRAK (a trademark of Black and Veatch) information management system used on the Project. POWRTRAK will be discussed in more detail later.
- Adtechs, with support from JGC, is responsible for the design of the radwaste facility. Adtechs designed, built and operates the Surrey radwaste facility.

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[1] The FOAKE Project has subsequently been completed in 1996.

- Adtechs, with support from JGC, is responsible for the design of the radwaste facility. Adtechs designed, built and operates the Surrey radwaste facility.
- Chicago Bridge and Iron is responsible for containment structural steel design.
- Shimizu is responsible for the majority of the reactor building civil design. Shimizu has major civil construction responsibility on the K-7 Project.
- Hitachi and Toshiba will provide consulting and engineering support to GE on the reactor building mechanical, control and electrical, and civil design.
- BATAN provides support to GE in the areas of mechanical and control and electrical systems design.
- CFE provides support to GE and Bechtel in the areas of mechanical and control and electrical systems design and piping stress analysis. CFE also provides support to Adtechs in the area of civil design.
- S3 Technologies and Technatom have provided support to GE in the areas of control room and man machine interface (MMI) design.
- ENSA provides support to GE in the area of reactor pressure vessel design, construction schedule, and cost estimate.
- UTE provides support to GE and Black and Veatch in the areas of mechanical and control and electrical systems design.

4.1.9.3 Entities involved (Taiwan)

Through a competitive bidding process, Taiwan Power Co. (TPC) selected the ABWR for its two unit Lungmen project. GE will design and provide the scope of supply for two 1,350 MWe ABWRs. The Lungmen project will also be supported by the GE Team including: Black & Veatch, Hitachi, Shimizu, Toshiba, and other US, Taiwanese, and international participants. Commercial operation for the two units is scheduled for 2003 and 2004.

4.1.9.4 Design status

The ABWR standard safety analysis report (SSAR) was completed in 1987, and is the basis for the US Nuclear Regulatory Commission (USNRC) licensing review. The SSAR describes the ABWR plant performance characteristics, methods and results of analyses, arrangement and layout drawings all in sufficient detail to enable the USNRC to complete its standard plant review. The ABWR SSAR includes information that forms the basis of construction verification and compliance reviews when the design is subsequently applied to actual projects.

Following TEPCO's decision to proceed with the two unit ABWR project at the Kashiwazaki-Kariwa site, construction of units 6 and 7 began in September 1991 and February 1992, respectively. Construction of the two units is on schedule, with significant milestones being achieved. The following table highlights some of the milestones of the project.

	K-6	K-7
Start of Construction	Sept. 1991	Feb. 1992
RPV Set	Aug. 1994	May. 1995
RPV Hydro	May. 1995	April 1996
Fuel Loading	Nov. 1995	Oct. 1996
Commercial Operation	Nov. 1996	1997[2]

At the completion of the ABWR FOAKE program in September 1996, the standardized US design will be approximately 65% complete, leaving only some of the detailed commodity engineering and site-specific engineering to be completed.

[2] Commercial operation was achieved in July 1997.

4.1.9.5 Research and development work

The ABWR design incorporates a number of advanced features which have undergone extensive R&D work. In many cases the advancements are adaptation taken from proven designs. Two prime examples of this are the reactor internal pumps (RIPs) and the fine motion control rod drives (FMCRDs).

The RIPs are an improved version of a European designed RIP of which nearly 100 are in operation in European nuclear plants. About 9 million pump-hours of successful operating experience has been accumulated, with some pumps having been in service since the mid-1970s. In addition, a test and evaluation program demonstrated that the RIP, as specifically designed for the ABWR, was highly reliable. This successful 2-year program included seismic tests, abnormal condition performance testing, life testing, thermal and vibration testing, and maintenance handling evaluations.

The FMCRD design is an improved version of similar drives that have been in operation in European BWRs since 1972. A total of 2,600 of these drives have accumulated about 35,000 drive years of experience during which time there have been no plant outages due to a FMCRD mechanical failure. Additionally, an extensive test and development program was undertaken to demonstrate the reliability of the FMCRD used in the ABWR, including an in-reactor test for over one year at an operating nuclear plant in the US

In addition to these, there were a significant number of additional test programs performed. The following list provides the general topics of a few of these additional test programs.

Vessel blowdown
Rapid flow coastdown
Core stability
Boron mixing
Control room human factors
Lower fuel spacer ΔP
RPV bottom head automated ISI
Reinforced concrete containment vessel
Containment horizontal vent
Reactor internals flow induced vibration

4.1.9.6 Licensing process

The overall goal of the ABWR program is the standard design certification by USNRC which requires meeting the requirements specified in 10CFR52, Early Site Permits; standard design certification; and combined licenses for nuclear power plants. Meeting these requirements requires the preparation by GE, and review by NRC of (1) the standard safety analysis report (SSAR); (2) A design specific probabilistic risk assessment (PRA) document; (3) proposed technical resolution of unresolved and all generic safety issues; and (4) proposed inspection, tests, analyses and acceptance criteria (ITAACs). Following the completion of review of the SSAR, the NRC will prepare the safety analysis report (SAR), which upon approval will result in the final design approval (FDA) following review (by the Advisory Committee on reactor safety, ACRS) and NRC acceptance of the ITAAC, PRA and all other NRC-requested data on safety-related technical issues provided by GE. The ABWR certification will then be issued following a rule making process.

4.1.9.7 Licensing status

In the fall of 1987, the USNRC began its technical review of the ABWR under the new 10CFR52 certification program. On July 13, 1994, the USNRC issued a final design approval (FDA) for GE's ABWR nuclear plant. The receipt of the FDA marked the culmination of the first phase of the overall

certification process. With receipt of the FDA, the ABWR became the only advanced plant in the world that has received regulatory approval in two countries, the US and Japan.

The second phase of the USNRC certification process, which includes public participation, is expected to result in design certification of the ABWR. The design will be approved for a 15 year period, with an option for renewal[3].

References

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24156-A10-SDD, "GE Advanced Boiling Water Reactor (ABWR) First-of-a-Kind Engineering Program System Design Descriptions," GE Nuclear Energy, San Jose, California, February 16, 1996.

23A6100 Rev. 7. "ABWR Standard Safety Analysis Report," GE Nuclear Energy, San Jose, California, July 1994.

Sawyer, C.D. "US ABWR Focus: Safety, operation and maintenance issues," Energy Horizons, GE Nuclear Energy, San Jose, California, May 1993.

[3] Subsequently Design Certification by the USNRC of the ABWR has been achieved in May 1997

4.2 APWR, MITSUBISHI, JAPAN/WESTINHOUSE, USA

4.2.1 Introduction

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

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

4.2.2 Description of the nuclear systems

4.2.2.1 Primary circuit and its main characteristics

Table 4.2-I shows a comparison of some major parameters between the APWR and an existing four-loop plant. The APWR is in the largest capacity class of LWRs in Japan and has adopted high performance steam generators and a highly efficient turbine with 52 in. (1320 mm) last stage blades. Various improvements have been incorparated in the reactor core so that operation with long fuel cycles is possible using low enriched fuel in order to reduce uranium requirements, and to provide increased flexibility for various application such as the use of MOX cores and high burn-up fuels.

Also critical equipment such as core internals and steam generators have been designed taking into account operational experience of aging on operating plants so that a high degree of reliability can be obtained. To ensure safety, the reliability of the equipment and systems has been increased, and highly advanced safety systems such as methods of providing assistance to the operations during abnormal events have been adopted. For instrumentation and control systems, the latest digital control technologies have been incorporated not only in the reactor control system but also in the reactor protection system, and also the latest electronics technologies to improve the man machine interface have been introduced in the main control room. In addition, in order to make the plant easier to maintain, a variety of improved technologies have been inconporated, thus improving the efficiency of periodical inspections and reducing exposure to employees.

4.2.2.2 Reactor core and fuel design

The reactor core, consisting of 257 improved 17 × 17 fuel assemblies, has a thermal output of approx. 4 127 MWt and has a relatively low power density. Considering the need to reduce fuel cycle costs and future needs for MOX reactor cores and high burn-up cores, a variety of improvements have been incorporated in the reactor core. Also the core has been designed so that it can use plutonium fuel with 1/3 or more MOX cores, and is flexible enough to use fuel with a burn-up of long operating cycles.

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

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

Table 4.2-I Major APWR parameters

	APWR	Existing four loop PWR
Electric power output	Approx. 1,420 MWe	1,180 MWe
Thermal power	Approx. 4,127 MWt	3,423 MWt
Fuel type	Improved type 17 × 17	17 × 17
Number of fuel assemblies	257	193
Fuel effective length	Approx. 3.7 m	Approx. 3.7 m
Total uranium inventory	Approx. 119 tonnes U	Approx. 89 tonnes U
Number of control rods	77	53
Reactor vessel	Appr. 5.2 m inner dia. & 14 m height	Appr. 4.4 m inner dia. & 13 m height
Steam generators	70F-1 type	52F type
Primary coolant pumps	100A type	93A-1 type
Primary system flow (m^3/h/loop)	Approx. 2.5×10^4	Approx. 2.0×10^4
Turbine	TC6F52	TC6F44
Containment	PCCV	PCCV
Engineered safety systems	Four trains of mechanical systems	Two trains
Refuelling water storage	Inside containment	Outside containment
Reactor protection system	Digital	Analog
Reactor control system	Digital	Digital
Main control room	Improved	Standard

4.2.2.3 Fuel handling and transfer systems

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

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

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

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

4.2.2.4 Primary components

Reactor pressure vessel

Although the reactor vessel inside diameter has been increased to 5.2 m in order to accomodate 257 fuel assemblies, the vessel is made with forged rings and has no longitudinal welds in the core area as is the case with the latest four-loop plant. The neutron irradiation of the steel opposite the core has been reduced to 1/2 or less than on present reactors by providing a neutron reflector, thus increasing the reliability of the reactor vessel.

Also, in order to reduce the susceptibility to corrosion cracking of the penetrations of the reactor vessel head, the nozzle material has been improved (from alloy 600 to heat treated alloy 690) and the primary coolant temperature in the top dome of the reactor vessel is designed to be reduced to reactor inlet temperature.

A comparison between the APWR and an existing four-loop plant is shown in Figure 4.2-1.

Reactor internals

The neutron reflector consisting of eight rings of stainless steel blocks not only reduces fuel cycle costs but also reduces the irradiation of the reactor vessel and core internals. By installing the neutron reflector, the neutron irradiation of the reactor vessel can be reduced to 1/2 or less compared to present reactors. On present reactors, the core baffle is a plate structure held together with 2000 or more bolts, whereas the new neutron reflector has a simple construction which does not use bolts in the core area.

In addition to the modification of the neutron reflector, the control rod guide tubes and other components have been improved to reduce vibration due to fluid flow and thus reduce wear of the lower support plate which stabilizes the flow inside the lower plenum and fretting of the control rods.

The improved core internals are now undergoing flow tests to check the validity of the design.

Steam generators

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

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

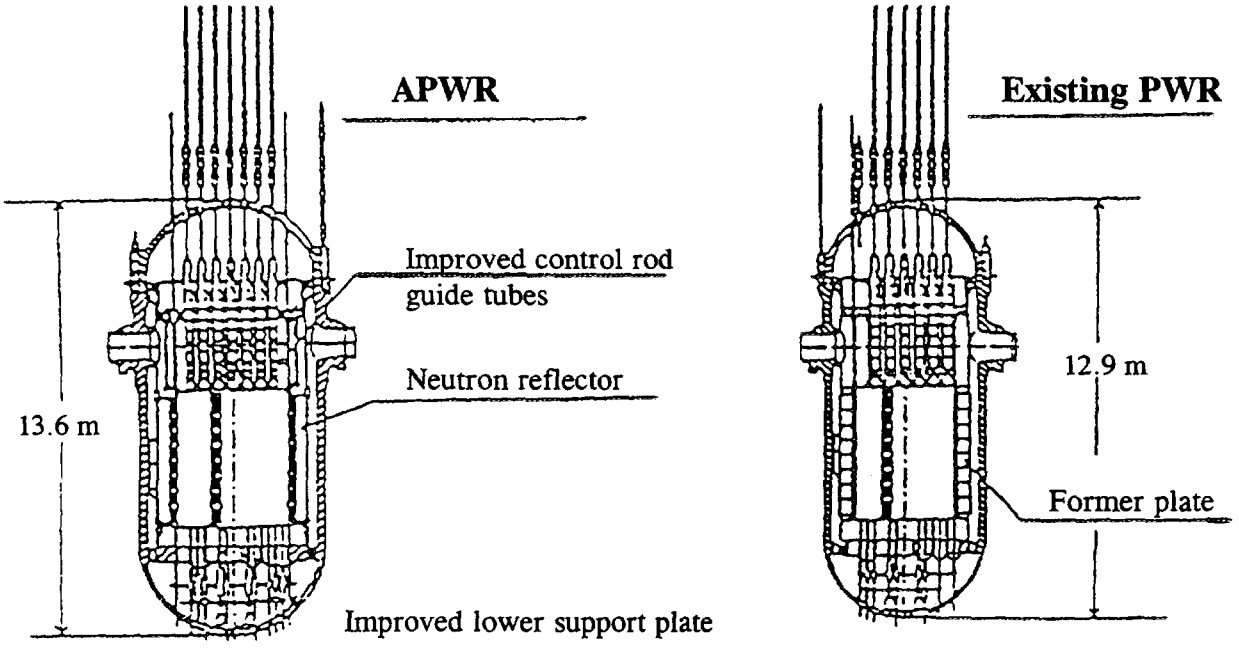


Figure 4.2-1 A comparison between the APWR and an existing PWR

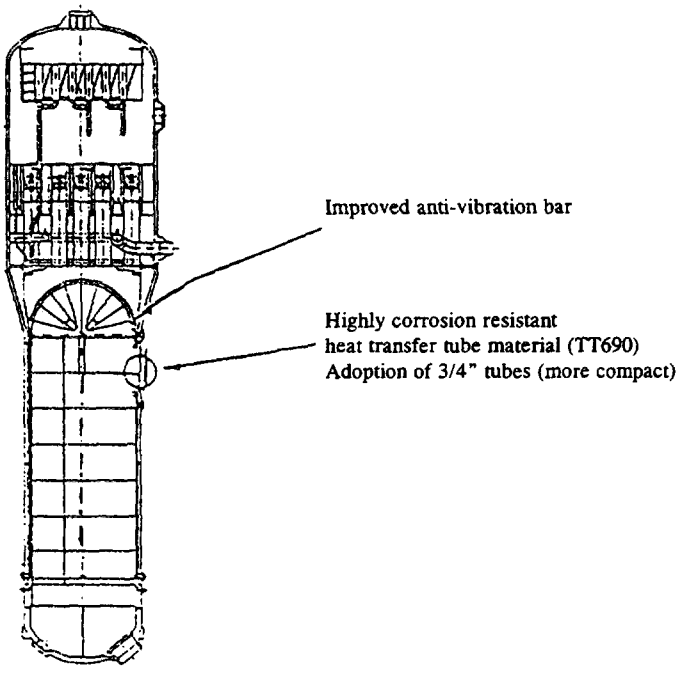


Figure 4.2-2 APWR - Steam generator

In addition, in order to make it easier to do maintenance and inspections, accessibility has been improved by increasing the diameter of the manholes and in other ways. Figure 4.2-2 shows a schematic drawing of the steam generator.

Pressurizer

No information provided.

Reactor coolant pumps

Because the primary coolant pump has to supply a flow approx. 10% larger than that of an existing primary coolant pump, a 100A type pump (60 Hz) which is larger in capacity than the existing 93A-1 type has been adopted. For the improved No.1 seals, heat-resisting O-rings are employed as well as ceramic material which has larger size and excellent durability, thus aiming at enhancement of the reliability during normal operation and complete failure of all AC power.

Main coolant lines

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

4.2.2.5 Reactor auxiliary systems

Chemical and volume control system

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

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

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

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

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

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

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

Reactivity control system

Reactivity control system is an equipment to inject emergency boron tank water (21 000 ppm B) installed at pump outlet via the containment direct injection nozzle (DVI) to the core, using two emergency boron pumps which take water from boron tank (commonly used with chemical and volume control system).

The major purpose of installation of this equipment is to inject boron into the reactor in the event of main steam pipe break accident in order to rapidly achieve the subcriticality of the core after occurrence of the accident.

4.2.2.6 Operating characteristics

The reactor is designed so that it can be operated automatically within the range of 15 to 100% of rated output by the reactor control system. Even in the low output range below 15%, the control rod control system can control the reactor automatically in the low power operating mode.

Usually the primary coolant average temperature is controlled to a programmed value which increases linearly with turbine output. When daily load following is used, however, the primary coolant average temperature is controlled to avariable reference temprature in order to reduce the amount of water that has to be processed.

The reactor control system is designed so that it can follow the following load change without causing a reactor trip.

- a 10% step load change (within the range of 15 to 100%)
- a 5% per min ramp load change (within the range of 15 to 100%)
- 100% load rejection

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

- Daily load-follow operation of 100%-50%-100% (14-1-8-1)
- Automatic frequency control or governor control to control system frequency over a load range of 5%.

4.2.3 Description of turbine generator plant system

4.2.3.1 Turbine generator plant

The high pressure turbine is a double flow turbine in which Rateux blades are used for the speed governing stage and reaction blading is used in all downstream stages.

Each low pressure turbine is a double flow turbine with reaction blading.

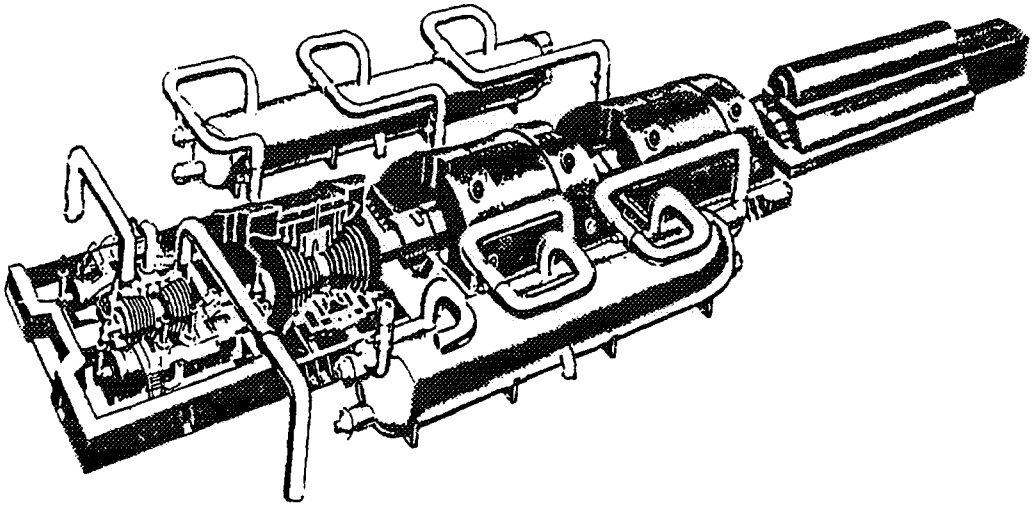


Figure 4.2-3 APWR - Turbine generator unit

The last stage blades are 52 in. (1320 mm) blades to increase the electric power output and efficiency. The high performance blades, multiple seals, and the new high performance guide vanes further increase the efficiency. Also the moisture extraction system has been improved to reduce erosion.

The 52 in. last stage blade has been subjected to vibration tests and actual load tests to demonstrate that the turbine will have a high performance and reliability. The performance has also been demonstrated on an actual machine.

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

The 1420 MWe electric generator which is of 4-pole type, has a larger rotor diameter than previous machines in order to increase the output. The rotor windings are cooled by hydrogen gas while water is used for cooling of the stator. The excitation is provided by a brushless static system.

Figure 4.2-3 shows a view of the turbine generator unit.

4.2.3.2 Condensate and feedwater systems

The feed heaters in the condensate and feedwater systems use six extraction stages in order to increase the efficiency. In particular, the deaerator is installed on the upstream side of the final high-pressure feed heater to obtain the benefits of a direct contact heat exchanger.

The condensate and feedwater systems consist of 3 trains of four stages of L.P. heaters, the deaerator and 2 trains of single stage HP heaters.

The feed heater tubing material is expected to be changed to stainless steel. This is to avoid corrosion of the LP feed heater tubes caused by ammonia which is added to control water purity.

The capacity of the pumps installed in the condensate and feedwater systems is 50% × 3 units (1 unit is spare). Even if one pump fails, full load operation can be maintained.

However, from the view point of economy, in the case of the main feedwater pumps, the spare pump is motor driven and its capacity is reduced to 40%, but operation under 100% load can be maintained if one normally operating pump fails.

4.2.3.3 Auxiliary systems

The auxiliary systems have the following additional features.

(a) To improve efficiency, the drains from each low pressure feed heater are collected in the next heater down stream.
(b) To further reduce the iron concentration in feed water, the water quality and equipment of the auxiliary systems have been improved.
(c) To simplify the systems and equipment, two large capacity circulating water pumps (50% capacity) have been installed in the circulating water system even though the plant output has been increased.

4.2.4 Instrumentation and control systems

4.2.4.1 Design concepts including control room

The main control room is provided with compact consoles on which CRTs and flat display panels are mounted. Conventional operating and monitoring devices such as switches lamps, indicators, and recorders have been eliminated.

The plant is operated by touch screen operations, and the monitoring information necessary for operation is displayed on the same screens that are used for operating the plant. Therefore the work load of the operators is reduced and the reliability of operation is increased.

On the wall of the main control room, a large display panel is installed to display the major monitoring parameters for normal and abnormal conditions of the entire plant. Thus the current status of the entire plant can be understood by everyone and communication between operators is improved.

4.2.4.2 Reactor protection system and other safety systems

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

The reactor protection system consists of four channels including the reactor trip breakers. Each channel is formed with multiple digital devices so as to provide redundant protection functions and to separate the reactor protection function from the other safety system operating functions.

The other safety systems consist of two trains. Each train has two sets of digital devices. To interface these systems with the auxiliary equipment in the plant, remote input/output devices arranged distributedly in the plant are connected to the host computer through optical fiber cables, thus reducing the volume of wiring.

The reactor protection system and other safety systems are provided with automatic test equipment so that periodical tests can be carried out fully automatically.

To achieve high reliability, the software used for the digital safety systems is modularized and simplified and a symbolic language was used in the design, Verification and Validation tests are to be carried out to the maximum extent possible.

In addition, as a countermeasure against accidental problems with the safety system software resulting any common cause failure, we plan to install a variety of automatic operating equipment for those safety functions which need urgent action and to install backup switches in the main control room for those critical safety functions for which there is sufficient time for operator action.

4.2.5 Electrical systems

4.2.5.1 Operational power supply systems

The operational AC power supply system can receive external power from the main power supply system and stand by power supply system. Power from the main power supply system comes through the main

transformer and plant transformers. When the plant is operating normally, the main generator is connected to the external power system and, when the main generator is stopped, it is disconnected from the external power system by the generator load break switch GLBS. Therefore, the operational AC power supply system can receive power through the plant transformers continuously whether the main generator is operating or not. If the main power supply system fails and the plant does not continue to operate independently, power will be received through the standby power supply system. Power from the standby power supply system is received through the standby transformer which has sufficient capacity to maintain the plant or hot standby conditions and enable it to be shut down safety.

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

In addition to the above AC power supply systems, an AC power system has been provided which can supplied from batteries in the event of an interruption or total loss of all AC power and an instrumentation and control power supply (consisting mainly of inverters) for supplying power to the instrumentation and control equipment which are mainly computer loads.

4.2.5.2 Safety related systems

The emergency power systems for supplying power to the operating power systems when an accident occurs in the plant or there is a loss of external power, include the emergency diesel generators and battery equipment. The emergency diesel generators will start automatically immediately after an accident occurs or external power is lost, and supply power to the emergency buses. The emergency power systems are made as redundant systems, and the safety of the plant can be secured with only one train of emergency power.

The DC power supply system can supply power to the instrumentation control power system during an instantaneous power failure, and also has sufficient capacity and to supply the switchgear which must operate following a loss of external power and to supply the excitation current for the diesel generators. Also it has sufficient capacity for maintaining the safety of the plant following a total failure of all AC power.

The bus configuration for the emergency power system is designed so that it is consistent with the configuration of the plant safety systems. As a result, the AC and DC power systems are divided into two trains to be consistent with the two trains and four sub-systems, and the instrumentation and control power system is divided into four trains to be consistent with the four channels.

4.2.6 Safety concept

4.2.6.1 Safety requirements and design philosophy

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

In existing plants, refilling of the reactor vessel and reflooding of the reactor core after a LOCA were made by both the accumulators and low presssure injection pumps. In APWR, however, advanced accumulators with two-stage discharge characteristics have been adopted and the present low pressure injection pumps have been eliminated to simplify the equipment and increase the functional reliability. Also, the refuelling water pit is installed inside the containment, thus eliminating the operation of changing the suction from the refuelling water tank to the containment recirculation sump which is needed during an accident on existing plants. A comparison between the ECCS of an existing plant and the APWR is shown in Figure 4.2-4.

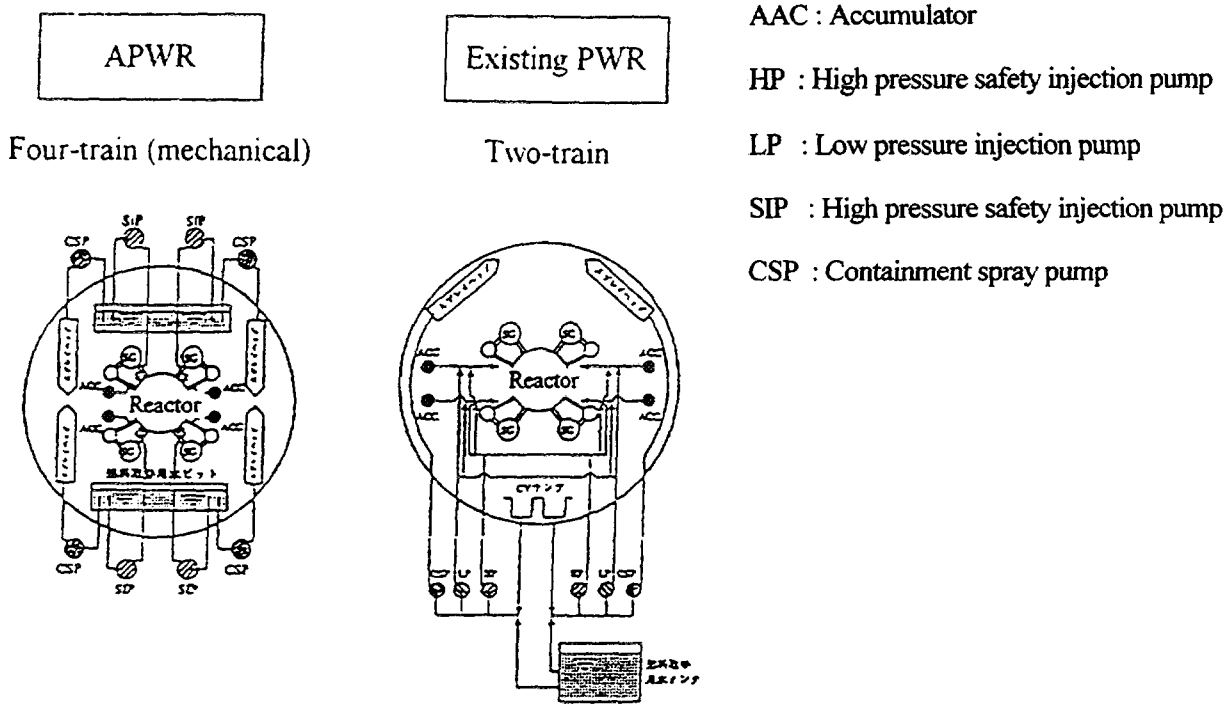


Fig. 4.2-4 APWR emergency core cooling systems compared with existing PWR

Also, we studied countermeasures for steam generator tube ruptures, reactivity-initiating accidents such as main steam line breaks, and severe accidents. As a result, it has been found that, as a whole, the APWR is expected to have a core damage probability of at least one order of magnitude lower than that of an existiting plant.

Deterministic design basis

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

Risk reduction

To further reduce the risk and provide increased protection, the reactor is designed to have a high degree of safety with simplification, economy, and ease of maintenance taken into account. Specifically, it is designed with the following design targets.

- The damage probability during power operation should be used as a quantitative index for the increase in safety. As a target, the probability should be reduced to about 1/10 th of that of the latest Japanese PWR.
- The core damage probability during shutdown should be approximately the same as the target for power operation.
- For greater protection, the containment failure probability (CFP) should be reduced to an appropriate level (to approx. 1/10 of the core damage probability as a target).
- The structure of the containment should be designed so that its functions as a target can be maintained for one day or longer during quasi-static pressurization following a severe accident. For premature failure modes caused by missiles and dynamic loads, countermeasures should be taken for equipment, etc.

Specially, these measures are concretely classified as follows.

(a) Countermeasures against core damage during power operation. Although a sufficiently low core damage probability can be achieved as a result of the increase in safety provided by the four sub-system safety systems, installation of the emergency water source inside the containment, etc., countermeasures against an interface LOCA and other events have been also taken which further reduce the risk.

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

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

External and internal hazards

No information provided.

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

Safety systems configuration

The primary system safety protection system consists of the residual heat removal system, safety injection system, and containment spray system which are installed in conventional nuclear power plants and the equipment has been integrated functionally.

The primary system safety protection system consists of four identical and independent mechanical sub-systems. Power is fed from two independent and redundant emergency power systems.

The basic configuration is as follows.

- Four sub-systems each having one safety injection pump, spray/residual heat removal pump, and spray/residual heat removal cooler
- One refuelling water pit installed inside the containment
- Four advanced accumulators tanks

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

The safety injection pump is connected so that it takes water from the refuelling water pit and feeds cooling water to the reactor vessel. The refuelling water pit is located at the bottom of the containment vessel.

The spray/residual heat removal pumps are used as residual heat removal pumps, and also used as containment spray pumps. The auxiliary feedwater system (AFWS) supplies the auxiliary feedwater required by the steam generator when the normal feedwater system is not available.

Except for the condensate pit, this system consists of two mechanical sub-systems. Each sub-systems consists of an electric auxiliary feedwater pump and a turbine driven auxiliary feedwater pump.

Safety injection systems

The emergency core cooling function of the primary system safety protection system is to feed sufficient cooling water into the core in a LOCA situation.

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

Immediately after the blow-down of the primary coolant during a LOCA caused by a large or medium sized break, the advanced accumulators are used to refill water into the reactor vessel lower plenum and downcomers, and to inject cooling water until the core is reflooded. At the start of injection, cooling water is injected with a large flow rate and then, when the water in an accumulator has dropped to a certain level, the flow damper switches over the flow to a smaller flow rate which provides an injection flow similar to that given by the safety injection pumps on current plants.

Containment spray system

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

When "P" signals to isolate the containment are initiated, the stop valves in the pump discharge lines are opened automatically. The spray/residual heat removal pumps take water from the refuelling water pit and supply it to the containment spray header located at the top of the containment through the spray/ residual heat removal coolers.

This system has also a residual heat removal function used to remove decay heat from the core in normal cooling of plants and refuelling.

In-containment refuelling water storage tank

The refuelling water pit is formed in a horse shoe shape, and is located at the bottom level of the containment.

It provides a continuous source of water for the safety injection pumps and spray/residual heat removal pumps. Therefore, it is not necessary to switch over from the refuelling water tank to the containment recirculation sump as must be done on current plants. During refuelling, the refuelling water pit is used also as a water source to fill the reactor cavity and fuel transfer system.

Emergency feedwater system

The auxiliary feedwater system, except for the condensate pit, consists of two mechanical subsystems. Each sub-system is provided with an electric auxiliary feedwater pump and a turbine driven auxiliary feedwater pump.

On receiving signals from the reactor protection system, the system starts feeding water automatically from the condensate pit to the steam generator. If the steam generator heat transfer tubes, main feedwater pipes, or main steam pipes are broken, the system isolates the auxiliary feedwater to the damaged steam generator automatically by auxiliary feedwater isolation signals.

When the auxiliary feedwater system has been started and the plant has been stabilized at hot standby conditions after an accident or transient, the auxiliary feedwater system can be used to cool the plant to the temperature at which the residual heat removal system can be put in service. When that temperature is reached, the residual heat removal system is started to bring the plant to cold shutdown, and the auxiliary feedwater system is stopped.

Residual heat removal system

The residual heat removal system of the primary system safety protection system removes decay heat from the core by taking water from the hot legs of the primary cooling system by the four spray/ residual heat removal pumps and returning the coolant to the cold legs of the primary cooling system through the four spray/residual heat removal coolers. The residual heat removal system has the capacity to cool the primary coolant temperature from 177 to 60°C within 20 hours after the reactor has been shut down.

4.2.6.3 Severe accidents (beyond design basis accidents)

Basic concept

In the preliminary design of the APWR a safety system with four sub-systems has been adopted, the refuelling water storage tank (RWST) has been located in the containment, advanced accumulators have been included, and the auxiliary feedwater system and auxiliary cooling water system/sea water system have been improved functionally. Thus, a high level of safety has already has been provided to ensure core integrity.

Regarding the interface LOCA which bypasses the containment, the corresponding parts of the piping in the residual heat removal system have been designed with a higher rating to prevent the interface LOCA from occurring since this type of accident can have very serious consequences to the environment.

Countermeasures during plant shutdowns

As safety enhancement during mid-loop operating mode, which are especially important among the countermeasures during plant shutdowns, there are countermeasures, such as adoption of RCS high water level operation, reinforcement of RCS water level indicators, automatic isolation of letdown line at low RCS water level, reinforcement of water injection functions during lowering of RCS water level, etc., in order to reduce the probability of core damage.

Also, as a precaution against the event of abnormal dilution of boric acid during the external power failure, interlocks to prevent boron dilution has been provided.

Mitigation of severe accidents

In an APWR plant, as shown in Section 4.2.6.1, the mitigation of the consequences of a severe accident is also considered from the view point of risk reduction and greater protection. Specifically, as shown below, countermeasures against those events that threaten the integrity of the containment vessel are under consideration.

As the countermeasures against debris dispersion, the reinforcement of depressurization function of the primary system and the improvement of RV cavity form are considered countermeasures against damage by quasi-static over pressure, the normal containment vessel air recirculation system and an alternative containment vessel spray supplied from the fire service water system can be used. These systems can be used to cool the containment vessel and reduce the pressure if the containment vessel spray system is not available.

As countermeasure against containment vessel damage due to hydrogen combustion, a hydrogen control system (ignitors) will be installed to control the hydrogen concentration.

To provide adequate cooling of molten debris ejected from the reactor vessel, sufficient floor space will be provided in the RV cavity and water will be injected into the cavity from the fire service water system. Also a 1 m thick protective layer of concrete will be provided so that the containment vessel boundary is not exposed directly to the debris. Thus the molten debris will be coolable, and erosion of the concrete and overheating of the containment vessel atmosphere can be prevented.

As countermeasures against the dispersion of debris, reinforcement of the primary system depressurization function and improvement of the reactor vessel cavity form are considered. It is also considered that the outlet from the RV cavity to the other containment vessel spaces should be constructed like a labyrinth.

Thus direct contact of the DCH or debris with the containment vessel boundary can be prevented.

4.2.7 Plant layout

The plant must be laid out so that the safety of the reactor facilities is not impaired, and the exposure dose around the plant is below a specified limit. Also separation of redundant trains, earthquake resistance, and maintenance of the safety system equipment must be considered to give an optimum arrangement.

4.2.7.1 Buildings and structures, including plot plan

Figure 4.2-5 shows the arrangement of an APWR plant. The standard arrangement is for a twin unit plant consisting of two reactor buildings and a common control building, waste building, and turbine building.

Design requirements

The buildings, structures, equipment, and pipes are classified into the following three classes: A, B, and C. The seismic design must be made according to the class concerned.

Class A: Those facilities which contain radioactive materials themselves or are associated directly with facilities which contain radioactive materials, and which may release radioactive materials outside the plant if they fail to function properly. Also, facilities which are required to prevent such releases and to reduce the effect of radioactive materials dispersed to the environment if an accident occurs, and those which have serious consequences for the plant.

Class B: Those which have relatively small effects on the plant compared to Class A.

Class C: Those for which safety standards equivalent to those of general industrial facilities are adequate and facilities not classified as A or B.

Based on the above classifications, the seismic design of the buildings is made by classifying the reactor building and control building into Class A, the waste building into Class B, and the turbine building into Class C.

To ensure safety against aircraft impacts, in principle, a site must be selected which is not close to an airport and air route if aircraft impacts are not to be considered in the design. However, if the location of the plant makes consideration of aircraft impacts unavoidable, the aircraft impact conditions must be determined properly, and necessary countermeasures must be taken.

The plant must be designed as follows for internal and external events such as jet aircraft, missiles, and fires.

Jet aircraft and missiles: The design must be made in such a way that the safety of the reactor is not impaired due to the effects of missiles and broken pipes.

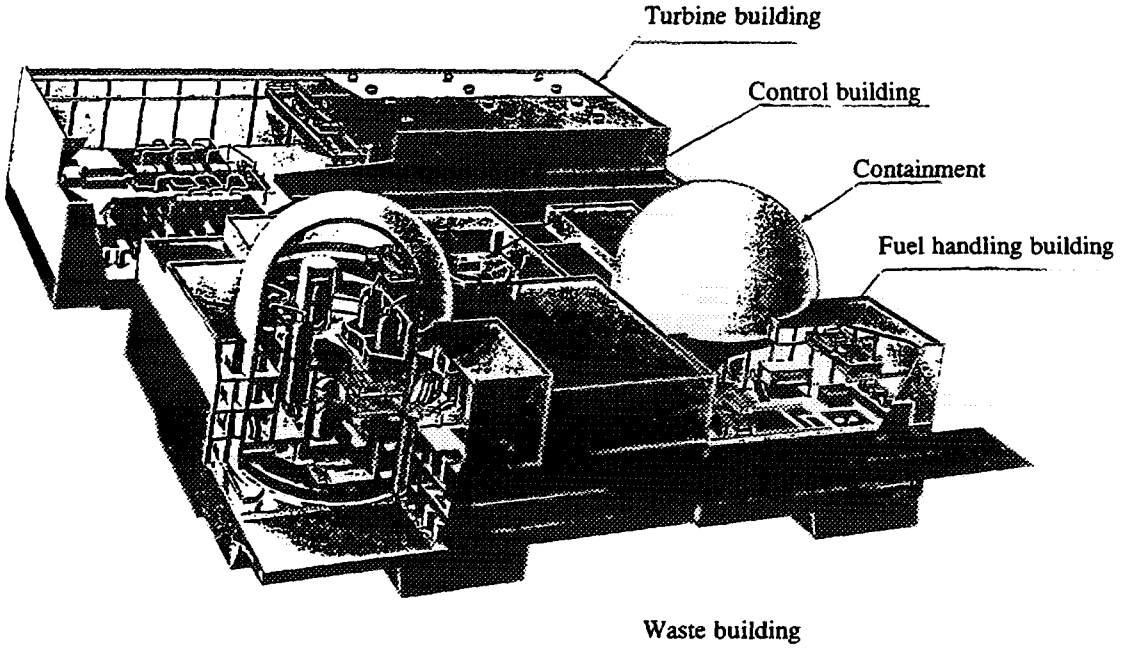


Figure 4.2-5 APWR - General view of buildings

Fire: To prevent the safety of reactor facilities from being impaired by fire, the plant must be designed using a proper combination of three general rules based on the Japanese "Guidance for Verification of Fire Protection of LWR Facilities for Power Generation."

(a) Prevention of fires
(b) Detection of fires and fire extinguishing
(c) Reduction of the effects of fires

In principle, the structures, systems, and equipment critical for safety must be so designed that the reactor facilities do not make common use of any one of them provided that, judging from the functions and construction, it is determined that the safety of the reactor may be impaired by such common use.

The reactor facilities must be arranged in the plant site of the plant so that the exposure dose to the general public in those areas outside the controlled areas and around the plant is below a specified limit. Furthermore, they must be sufficiently far from the site boundary so that the exposure dose received in areas outside the site during severe accidents and hypothetical accidents is adequately below the target dose indicated in the Japanese "Guidance for Verification of Reactor Siting."

The interior of the plant must be divided into zones according to the radiation levels, and suitable radiation shielding must be provided.

4.2.7.2 Reactor building

The reactor building consists of the reactor containment facility and the peripheral buildings in which the fuel handling systems and associated systems are installed. Figure 4.2-6 shows a cross-section of the reactor building.

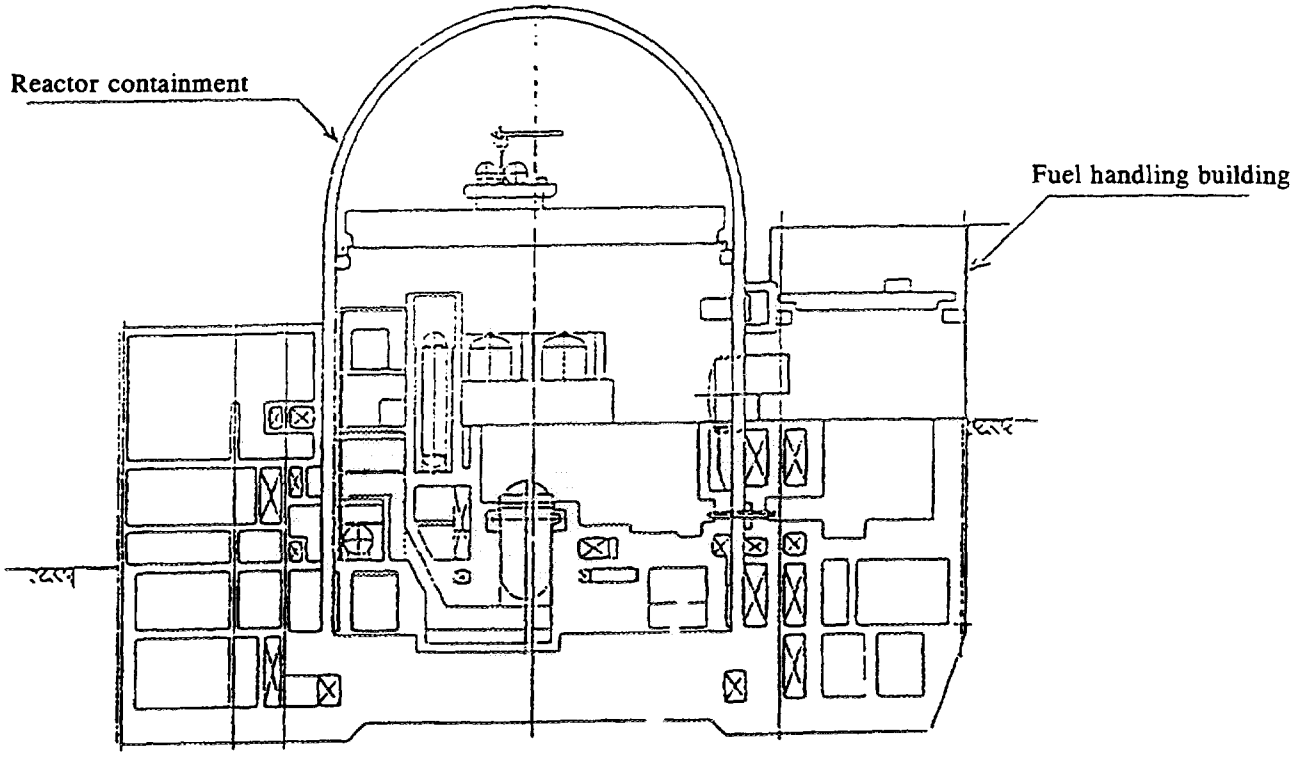


Figure 4.2-6 APWR - Reactor building cross-section

4.2.7.3 Containment

The containment is part of the reactor containment facility and includes the internal concrete and the annulus. The reactor containment facility is part of the engineered safety systems which include the emergency core cooling system, the containment spray system, and the annulus air purification system.

The containment system is designed to suppress or prevent the possible dispersion of the large quantities of radioactive materials which would be released if extensive fuel failures were to occur in the reactor resulting from damage or failure of the reactor facilities such as the primary cooling system, main steam system, and feedwater system.

The leakage preventing function of the containment is provided by a 6.4 mm thick steel liner on the inner surface while the pressure withstanding function is provided by the concrete structure. An enclosed space (annulus) is formed around the lower part of the containment shell to provide a double containment and the containment penetrations for pipes, cables, ducts, and air locks pass through the annulus.

The containment is designed so that the leak-rate is less than 0.1% per day of the weight of air in the containment at a pressure of 0.9 × maximum design air pressure at normal temperatures.

The containment is provided so that the general public will not be affected by radiation if it leaks at this leak-rate even if the facilities related to the primary coolant system fail or are damaged. Therefore, severe accidents must also be carefully considred to ensure the integrity of the containment.

In current-day plants, the refuelling water which is the water supply used after an accident, was stored in a tank outside the containment. In an APWR, however, in order to avoid a failure to switch over the water source from the tank to the recirculation sump inside the containment, the refuelling water is stored in a pit inside the containment.

Also, a proper space has been provided below the reactor vessel so that debris will not concentrate in one location if a hypothetical ejection of molten debris occurs, and the space is shaped to catch the debris easily to prevent it from being splashed, as far as possible, into the general spaces of the containment.

Although the containment is designed to withstand the maximum design pressure and temperature determined from the design basis events, there will be no excessive leakage under the pressure and temperature conditions expected during severe accidents. This must be checked to make sure.

4.2.7.4 Turbine building

The turbine generator, condensate and feedwater system auxiliary equipment, and other equipment are installed in the turbine buidling. The turbine generator is arranged with its axis in line with the reactor.

The foundation of the turbine building is made of concrete to reduce the thickness of the mat.

The floor of the turbine building below ground level is made of concrete, and the floors above ground are steel structures which are designed to withstand all loads including the load of the overhead travelling crane.

The turbine generator systems are arranged so that the space can be utilized effectively not only during the construction of the plant but also during operation and periodical inspections.

Suitable spaces have been provided for inspection access, transportation of tools for inspections and maintenance, and disassembly in a way that reduces the volume of the building.

4.2.7.5 Other buildings

The buildings and systems have been arranged so as to optimize the relation between the systems, separation of safety system equipment, seismic resistance, maintenance, etc.

Fuel handling building

The fuel handling building is the part of the reactor building which contains the spent fuel pool, the cask pit, the fuel inspection pit and associated equipment.

Control building

The control building which is common to both units, contains mainly the main control room, electrical equipment and access control equipment.

Waste building

The waste building which is common to both units, mainly houses the radioactive waste treatment systems.

4.2.8 Technical data

General plant data		
Power plant output, gross	1 420	MWe
Power plant output, net		MWe
Reactor thermal output	4 127	MWt
Power plant efficiency, net		%
Cooling water temperature		°C
Nuclear steam supply system		
Number of coolant loops	4	
Primary circuit volume, including pressuriser		m^3
Steam flow rate at nominal conditions		kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	-/-	°C/MPa
Feedwater temperature/pressure		°C/MPa
Reactor coolant system		
Primary coolant flow rate [27.76 m^3/s]		kg/s
Reactor operating pressure	15.7	MPa
Coolant inlet temperature, at RPV inlet		°C
Coolant outlet temperature, at RPV outlet		°C
Mean temperature rise across core		°C
Reactor core		
Active core height	3.7	m
Equivalent core diameter		m
Heat transfer surface in the core		m^2
Fuel inventory	119	t U
Average linear heat rate		kW/m
Average fuel power density	34.7	kW/kg U
Average core power density (volumetric)		kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		

Fuel material	Sintered UO_2	
Fuel assembly total length	3 900	mm
Rod array	square, 17×17	
Number of fuel assemblies	257	
Number of fuel rods/assembly		
Number of control rod guide tubes		
Number of spacers		
Enrichment (range) of first core		Wt%
Enrichment of reload fuel at equilibrium core		Wt%
Operating cycle length (fuel cycle length)		months
Average discharge burnup of fuel		MWd/t
Cladding tube material	Zr 4	
Cladding tube wall thickness		mm
Outer diameter of fuel rods		mm
Overall weight of assembly		kg
Active length of fuel rods		mm
Burnable absorber, strategy/material		
Number of control rods assemblies (RRC)	77	
Number of grey control rods assemblies (GRC)		
Number of water displacer rods assemblies (WDR)		
Absorber rods per control assembly		
Absorber material: RCC	Ag-In-Cd + B_4C	
GRC	Stainless steel	
WDR	Zircaloy	
Drive mechanism: RCC and GRC	Magnetic jack	
WDR	Hydraulic piston	
Positioning rate		steps/min [or mm/s]
Soluble neutron absorber		
Reactor pressure vessel		
Cylindrical shell inner diameter	5 200	mm
Wall thickness of cylindrical shell		mm
Total height	14 000	mm
Base material: cylindrical shell		
RPV head		
Design pressure/temperature	/	MPa/°C
Transport weight (lower part)		t
RPV head		t

Steam generators

Type	70F-1, U-tube heat exchanger	
Number	4	
Heat transfer surface		m^2
Number of heat exchanger tubes		
Tube dimensions	19	mm
Maximum outer diameter		mm
Total height		mm
Transport weight		t
Shell and tube sheet material		
Tube material	TT 690 alloy	

Reactor coolant pump

Type	100A, Single-stage, centrifugal pump	
Number	4	
Design pressure/temperature		MPa/°C
Design flow rate (at operating conditions) [6.94 m^3/s]		kg/s
Pump head		
Power demand at coupling, cold/hot		kW
Pump casing material		
Pump speed		rpm

Pressuriser

Total volume		m^3
Steam volume: full power/zero power		m^3
Design pressure/temperature	/	MPa/°C
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter		mm
Total height		mm
Material		
Transport weight		t

Pressuriser relief tank

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Dry,	
Overall form (spherical/cyl.)	cylindrical, steel	
Dimensions (diameter/height)		m
Free volume		m^3
Design pressure/temperature (DBEs)	/	kPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate	<0.1	vol%/day
Is secondary containment provided?		

Reactor auxiliary systems

Reactor water cleanup,	capacity	kg/s
	filter type	
Residual heat removal,	at high pressure	kg/s
	at low pressure	kg/s
Coolant injection,	at high pressure	kg/s
	at low pressure	kg/s

Power supply systems

Main transformer,	rated voltage	kV
	rated capacity	MVA
Plant transformers,	rated voltage	kV
	rated capacity	MVA
Start-up transformer	rated voltage	kV
	rated capacity	MVA
Medium voltage busbars (6 kV or 10 kV)	6.6	kV
Number of low voltage busbar systems		
Standby diesel generating units: number		
rated power		MW
Number of diesel-backed busbar systems		
Voltage level of these		V ac
Number of DC distributions		
Voltage level of these		V dc
Number of battery-backed busbar systems		
Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	TC6F52	
Number of turbine sections per unit (e.g. HP/LP/LP)		
Turbine speed		rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/°C

Generator

Type	
Rated power	MVA
Active power	MW
Voltage	kV
Frequency	Hz
Total generator mass	t
Overall length of generator	m

Condenser

Type	
Number of tubes	
Heat transfer area	m^2
Cooling water flow rate	m^3/s
Cooling water temperature	°C
Condenser pressure	hPa

Condensate pumps

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

Condensate clean-up system

Full flow/part flow	
Filter type	

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/ °C

Feedwater pumps

Number	
Flow rate	kg/s
Pump head	
Feedwater temperature	°C
Pump speed	rpm

Condensate and feedwater heaters

Number of heating stages	
Redundancies	

4.2.9 Project status and planned schedule

No information provided

References

No information provided

4.3 BWR 90, ABB ATOM, SWEDEN

4.3.1 Introduction

The BWR 90 standard plant design of ABB Atom represents an "evolution" of the design of its successful predecessor, the BWR 75, with a number of design modifications, improvements and supplements that address new licensing requirements and aim at meeting utility needs for increased public safety, investment protection, lowered cost, and ease of operation and maintenance.

The BWR 90 design is characterized by the use of internal recirculation pumps, fine motion control rod drives, a prestressed concrete containment, and extensive redundancy and separation of safety related systems in the same way as the BWR 75 design that was developed in the 1970s. The modifications are mostly moderate and they have been made to adapt to updating technologies, new safety requirements and to achieve cost savings.

There is one easily distinguishable departure from previous designs, however; the containment arrangement. In the new concept the connections between the drywell and the condensation pool in the wetwell are accomplished in a quite different way, and design measures to cope with a "degraded core" accident have been incorporated (by provision of a core catcher arrangement and filtered venting for the containment) in order to ensure that public and environment will be protected even in the event of a degraded core accident situation.

The BWR 75 design included two standard sizes, with nominal thermal power of 2000 MWth and 3000 MWth, respectively. During the 1980s, the BWR 75 plants have been successfully uprated by 8-9%, taking advantage of improvements in fuel technology. These upratings required only minor modifications to plant systems and equipment and were carried out at a very low cost. The BWR 90 originally also had two standard sizes, closely corresponding to the BWR 75 sizes - with nominal thermal power of 2350 and 3300 MWth, respectively. These standard sizes have later been supplemented by a larger unit - with a nominal thermal power of 3800 MWth - taking advantage of the margins that are gained by utilization of the new generation of ABB Atom BWR fuel. The size described in the following is the 3300 MWth size.

As noted above, the BWR 90 is not a new reactor concept; it is based on the design, construction, commissioning and operation of a number of BWR 75 plants in Finland and Sweden, and it has been developed by making specific changes to an established reference design, the Forsmark 3 and Oskarshamn 3 power plants, with a strong emphasis on maintaining "proven design" features unless changes would yield improvements and simplifications.

The operating records of the company's BWR plants show high plant availability and power production reliability, and low occupational radiation exposure. A basis for such achievements is a good basic plant design; not only with respect to systems performance and component reliability, but also a design which from the beginning has taken the needs for maintenance and service into consideration. The operating utility obviously has a profound influence on the plant performance, but even a proficient utility will most likely fail to achieve good results, if the plant design is not good enough.

A "suitable" plant design involves many different aspects - the design of various systems, choice of materials and components, their installation, radiation shielding, accessibility to components, transport routes, proper routing of ventilation air, general building arrangement, etc. The end result will always represent a compromise between a number of concerns, and in this context, a co-operation with the Finnish utility Teollisuuden Voima Oy (TVO), with its feedback of practical experience, has been of great value for the development of BWR 90.

In line with the strong preferences given to "proven design" features and solutions in the development work, - an approach that was firmly supported by TVO, - it is easily concluded that the design of the reactor has changed very little, and that the nuclear island as a whole has not been changed much.

Some of the special features of the BWR 90 are reviewed briefly below. The description reflects the design which was offered to Finland as one of the contenders for the fifth Finnish nuclear power plant project.

4.3.2 Description of the nuclear systems

4.3.2.1 Primary circuit and its main characteristics

The general reactor pressure vessel arrangement is very much the same as in the Forsmark 3 and Oskarshamn 3 plants in Sweden - with steam and feedwater lines connected to the upper portion of the vessel and with housings for the recirculation pump motors integrated with the bottom of the pressure vessel.

In BWR plants, the heat generated in the reactor core is utilised directly for "steam generation" inside the reactor vessel; the steam develops as small "bubbles" (void) entrained in the core coolant, is separated from the coolant flow in the steam separators that are located on top of the moderator tank cover, and "dried" in a steam dryer arrangement at the top of the reactor vessel. The removal of water droplets in the steam dryers - to attain an acceptable moisture content in the steam - serves two purposes; it minimises the risks of excessive erosion of turbine blades in the first stage, and reduces the carry-over from the reactor water to the turbine systems of more long-lived radioactive products. The nitrogen and oxygen isotopes N-16 and O-19 are very short-lived and do not represent any practical constraints with respect to service and maintenance on turbine systems.

The reactor power level is set by withdrawing (to increase) or inserting (to decrease) control rods from the core region, but control rod patterns are not changed frequently to achieve power level adjustments; normally, control rod manoeuvering is used only to compensate for reactivity burnup during the operating cycle.

As noted above, the BWR is characterized by the presence of void in the core coolant during normal operation, and this yields a strong feedback of coolant flow rate; an increased flow rate results in a decreased void content and a subsequent increase in reactor power. Therefore, the reactor power can be easily controlled by means of adjusting the recirculation pump flow rate.

The recirculation system is based on the use of internal glandless pumps which provides means for an accurate control of the reactor power, and eliminates large break LOCAs below the core level. It also provides for a recirculation flow path with lowered pressure losses; i.e. the natural circulation flow rate increases and the core cooling capability at loss of forced circulation improves significantly.

The "dried" steam is conveyed from the RPV to the turbine plant through four steam lines. The steam lines connect to nozzles with built in "flow limiters", evenly distributed along the vessel circumference; own medium operated isolation valves are provided on the inside and outside of the containment wall, the outer valve is equipped also with a motor operated actuator to ensure leaktightness after closure.

The feedwater lines enter the containment via two lines, each with inner and outer isolation valves, splitting up into four lines adjacent to the RPV for connection to four nozzles, at "mid-height" of the vessel. The nozzles and the internal removable feedwater distributors are of a special ABB Atom design that ensures a "thermal sleeve" protection against the "cold" feedwater for the RPV wall, and an efficient distribution into the downcomer. The feedwater flow rate is adjusted to

match the steam flow rate from the vessel, to keep the water level within close limits, by speed control of the feedwater pumps at high power operation, but valve arrangements enable flow rate control also at low reactor power levels; when operating at low power, the feedwater flow is routed via smaller nozzles that can easier withstand thermal transients.

The RPV is provided with a pressure relief system which consists of 16 safety (relief) valves connected evenly onto the four steam lines, with blowdown pipes leading down into the condensation pool. The safety (relief) valves are own medium operated valves, each being controlled by two pilot valves, one pressure activated and one electrically controlled; this means that actuation can be initiated in a controlled way by pressure monitoring equipment, to avoid overpressurization or to achieve depressurization. In addition, control valves are provided downstream two of the safety valves, in order to enable proper pressure control of the reactor also in the event of isolation (loss of the turbine condenser function). Downstream each of the valves there is also a "forced closure" valve which permits closing (one at a time) any inadvertantly opened safety/relief valve; a stuck open relief valve is no major safety concern.

In comparison with its predecessor, the primary circuit of the BWR 90 incorporates improvements related to the design of the reactor pressure vessel proper, to the reactor core and the reactor internals, to the recirculation pumps and their control systems, to reactor auxiliary systems, and to the installation and arrangement of the reactor primary system in the containment.

4.3.2.2 Reactor core and fuel design

The reactor core is a typical ABB Atom BWR core, made up of 700 fuel assemblies of the SVEA-100 type. In the BWR 75, the core design was based on traditional 8×8 fuel assemblies with a rod diameter of about 11 mm; the SVEA fuel assemblies introduced 4×4 subassemblies with an internal cruciform water gap between them. This water gap significantly improves moderation and reduces local power and burnup peaking factors. It also contributes to a mechanically favourable fuel channel structure with a very low creep deformation and a minimum amount of neutron absorbing Zircaloy. Advanced utilization of burnable absorber material (Gd_2O_3), axially and radially graded, in the fuel made it possible to achieve good axial and radial power distribution with low peaking factors, and good operating margins.

The introduction of the SVEA-100 fuel represents a further improvement; the 4×4 sub-assemblies are replaced by 5×5 subassemblies with thinner fuel rods (about 9 mm in diameter). This yields a significant increase in total fuel rod length and cladding surface and a corresponding decrease in average heat rate and surface heat flux. The increased operating margins can be used to increase average core power, to improve total neutron economy, or for a combination thereof, and improved thermal-hydraulic stability. For the BWR 90, a portion of the increased margins has been taken into account to raise the power level of the reactor.

A group of four fuel assemblies, surrounding a cruciform control rod, makes up a core module unit. The control rod blades and control rod drives for the BWR 90 are of a well-proven design. The cruciform rod is based on solid steel blades that are welded together. Holes filled with B_4C as neutron absorber are drilled horizontally in the blades. In the top of the rod, the absorber consists of Hafnium which makes the rod tip more "grey" and provides for a long service life.

The control rod drives (CRDs) utilize two separate drive mechanisms, one electro-mechanical and one hydraulic. The former is used for normal, continuous fine motion of the control rod - for burnup compensation or for adjustment of the power distribution - whereas the latter is used for rapid control rod insertion (scram). The control rods are divided up into scram groups; each group is equipped with its own scram module, consisting of a scram tank, piping and valve. A total of 18 such scram groups are provided, comprising 8 to 10 rods. The rods belonging to any one group are distributed over the core in such a way that the reactivity interference between them is virtually negligible. The consequence of a failure in one scram group is therefore no more serious than

sticking of a single rod. The scram signal also initiates a rapid run-back of the recirculation pumps and a continuous insertion of all rods by the electro-mechanical drives, as a back-up to the hydraulic insertion. The latter function is also initiated by an independent scram back-up circuit of the reactor protection system.

The diversified means of control rod actuation and insertion (together with a generous reactor pressure relief capacity) provide, in combination with a capability of rapid reduction in the recirculation flow rate (recirculation pump run-back), an adequate countermeasure against ATWS (anticipated transient without scram).

4.3.2.3 Fuel handling and transfer systems

The fuel handling and refuelling operations are quite similar to those in previous plant designs. The fuel pools are located in the reactor service room on top of the containment, one on each side of a central reactor pool enclosing the containment vessel dome. During plant operation, the reactor pool is filled with water to provide adequate radiation shielding, but when shutting down for refuelling, or for other purposes requiring a reactor vessel opening, the central portion of the reactor pool is drained from water to provide access to the containment dome and the reactor pressure vessel head. When these have been removed, the reactor pool is filled with water again. Then the reactor internals are lifted out of the vessel, using the overhead crane, and placed in storage positions in the pool. Now the core is accessible for fuel handling, by means of a traditional refuelling machine in the reactor service room. Spent fuel assemblies are brought up to the reactor pool and transported to storage racks in an adjacent fuel storage pool via a transport gate; in typical 12 month operation cycles 20-25% of the fuel assemblies are replaced during the refuelling.

Fresh fuel is delivered to the plant during the operation season and brought up to a fresh fuel storage in the reactor building located below the reactor service room, via a transport shaft. Prior to start of the refuelling operations, the fresh fuel is transferred to the fuel pool in dedicated storage rack positions, via a separate transport opening, using a service room crane. During the refuelling operations, the transportation of fresh fuel assemblies takes place by the "return trip" of the refuelling machine - to the reactor pool and down to the core and their proper position.

Spent fuel is stored in the spent fuel pool for a number of years before it is transported away to an "away from reactor" storage facility in special transport containers, via the main transport shaft of the reactor building. The fuel pools are typically provided with storage racks that have sufficient capacity for storage of one complete core plus the spent fuel arising from seven years of normal operation.

4.3.2.4 Primary components

Reactor pressure vessel

The general reactor pressure vessel arrangement is the same as in the Forsmark 3 and Oskarshamn 3 plants; with steam and feedwater lines connected to the upper portion of the vessel and with the recirculation pump motor housings integrated with the pressure vessel at the lower portion. The steel vessel proper has been modified slightly, however. The cylindrical portion is made up of cylindrical forgings in the same way as in the Forsmark 3 and Oskarshamn 3 plants; this eliminates the longitudinal welds. The bottom portion is redesigned in such a way that large sections of it can be made by forging; the number of welds is reduced significantly. This reduction in number of welds is important since it reduces the amount of in-service inspection to be carried out during the refuelling outage. The reactor vessel length is 20.9 m and the width is 6.4 m.

Reactor internals

The moderator tank and the core support plate arrangement correspond closely to the BWR 75 design; this applies also to the moderator tank cover. The steam separator units on top of the cover have been improved - as well as the steam driers in the upper portion of the vessel - in order

to ensure low moisture content in the steam at the increased power output level; the basic arrangement of the units is just the same as in previous plants.

Reactor recirculation pumps

The recirculation system is based on the use of internal glandless pumps driven by wet, asynchronous motors, supplied individually with "variable frequency - variable voltage" power from frequency converters. This type of pumps has been operating reliably in ABB Atom reactors (for more than three million operating hours) since 1978.[1]

In BWR plants, the reactor power is easily controlled by means of the recirculation pump flow rate. Normally, an upper level of reactor power is established by means of control rod manoeuvering until a certain control rod pattern in the core has been attained, and then adjustments of the recirculation flow rate are utilized to control the power level. A BWR is characterized by the presence of void in the core coolant during normal operation, and this yields a strong feedback of coolant flow rate; an increased flow rate results in a decreased void content and a subsequent increase in reactor power. Therefore, the internal pumps provide means for rapid and accurate power control in the high power (or normal operating) range, and they are also advantageous for load following purposes.

The internal recirculation pumps are provided with more than 10% excess flow rate capacity, which allows xenon override, and the fine motion control rod drives and the grey-tipped control blades allow control rod movements at full power. The excess pump capacity is utilized for hydraulic spectral shift operation; the core coolant flow is increased towards the end of the operating cycle. The built-in "redundancy" also implies that the reactor can be operated at full power even if one pump should fail.

4.3.2.5 Reactor auxiliary systems

A shut-down cooling system with one high pressure and two low pressure loops is provided for the "normal decay heat removal" function when the reactor is shut down to cold conditions. A reactor water clean-up system (RWCU), with a radial flow type of deep bed filters, heat exchangers (one of regenerative type), and pumps, draws water from the shut-down cooling system nozzles and returns it as purge flows through the control rod drives and the recirculation pump housings or discharges directly into the vessel.

Other auxiliary systems serve to cool and clean the water in the condensation pool in the containment wetwell and the water in the reactor service and spent fuel storage pools on top of the containment structure.

The main development objective related to the reactor auxiliary systems was to evaluate possible simplification of their design in order to achieve cost reductions and more straight-forward operation. The reactor water clean-up system can be taken as an example on this review. In previous plants, a certain flow rate of reactor water, a percentage of the full power feedwater flow rate, was continuously passed through the RWCU filters, and a forced flow mode (at twice the flow rate) was initiated when needed. In BWR 90, the RWCU operation is controlled by the water chemistry in the reactor; during normal full power operation cleanup needs are limited and only a small reactor water flow is passed through the RWCU, but whenever measurements show a need, the RWCU is taken into operation at full capacity. This reduces the heat losses etc., and therefore yields "cost reductions".

[1] Such internal pumps are adopted also by other BWR vendors, in the ABWR plants.

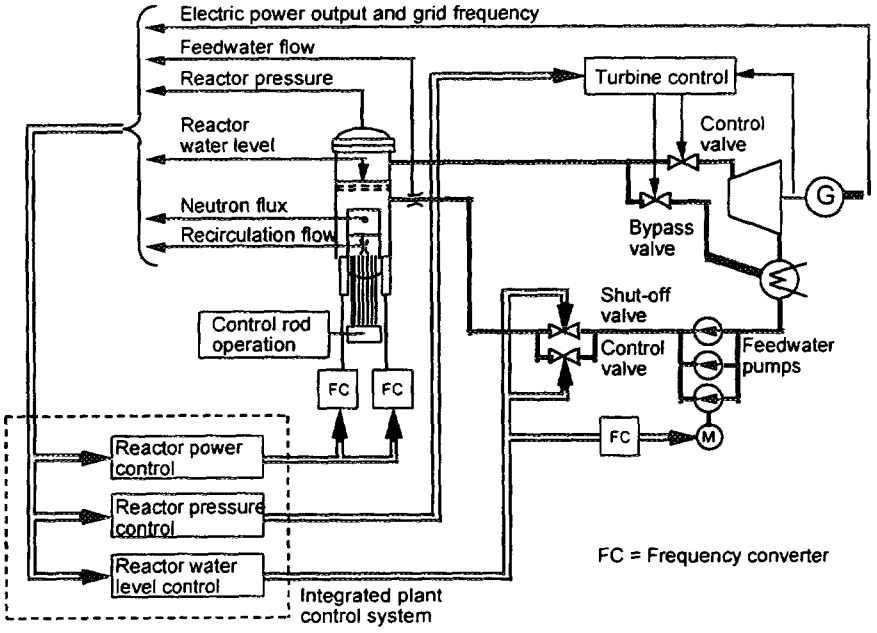


Figure 4.3-1 BWR 90 - Interrelationship between reactor control systems

4.3.2.6 Operating characteristics

A schematic overview of the interrelationship between the reactor control systems with respect to operation is shown in Figure 4.3-1.

The internal recirculation pumps provide means for rapid and accurate power control and are advantageous also for load following purposes. The BWR 90 plant is characterized by a capability to accept a 10% step change in power with an equivalent time constant of down to 5 seconds, and ramp load changes of 20% per minute is accepted. In the high power range, between 70 and 100% of nominal power, daily variations with the above change rate can be accommodated without restrictions; for wider power variations, the extended range is achieved by control rod pattern adjustments - at a rate of change of 1-2% per minute. Daily load following in a 100-40-100% cycle with (1 -) 2 hour ramps can be accommodated.

The plant power control system is provided with a frequency control mode in which a certain portion of the output (typically $\pm 2\%$) is assigned for participation in the grid frequency control strategy - with an adjustable deadband and an adjustable amplification factor. The settings for the power control, and the control modes, are normally selected by the operator in charge (in the control room), but the plant may also be connected up to a grid dispatch centre - for remote power control from this centre in accordance with the anticipated needs of the grid.

With respect to operating characteristics it may also be noted that the plant is designed to withstand a full load rejection without being tripped; the plant will shift to house load operation, being prepared for a return to normal operation. The plant is further designed to avoid a reactor trip in the event of turbine trips, as long as the turbine condenser remains available for steam dumping, and to withstand certain grid voltage disturbances (voltage drops due to short circuits and other electrical faults on the grid) without being disconnected from the grid.

In the context of operating reliability, a particular design prerequisite may be highlighted; a malfunction or failure of a single component is allowed to cause a temporary disturbance but it should not lead to a trip of the reactor. To this end, significant redundancies in process components, control systems and related power supplies are found in all functions of major importance to the plant operation.

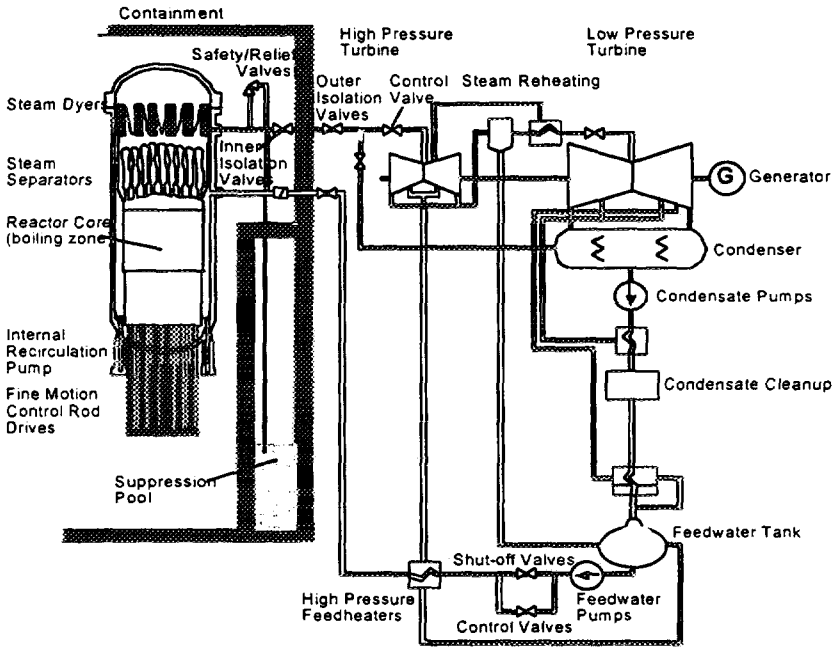


Figure 4.3-2 BWR 90 - The power conversion process

4.3.3 Description of turbine generator plant systems

4.3.3.1 Turbine generator plant

The power conversion process is depicted on Figure 4.3-2.

The reference turbine plant design of the BWR 90 is similar to that of modern, existing BWR plants. The nominal power output of the turbine unit will be 1100-1240 MWe depending on the site conditions, in particular with respect to the temperature of the circulating water.

The saturated steam from the reactor vessel is conveyed to the admission valves of the high pressure cylinder via the four steam lines. After expansion through the HP unit, the steam passes through a steam moisture separator unit and a steam reheater, on its way to the admission valves of the three low pressure turbine cylinders. A special "steam bypass" line, from the steam lines to an intermediate extraction point, provides a means for temporary increases of the steam flow to the turbine cylinders - for power control purposes.

A full-capacity steam bypass system is also provided to enable dumping the full nominal steam flow directly to the main turbine condenser in the event of certain disturbances, in order to avoid pressure surges, and corresponding power peaks, in the reactor.

The generator is a two-pole type turbo generator, designed for continuous operation with hydrogen as the cooling medium for the rotor and water as the cooling medium for the stator windings. Its rotor is directly coupled to the turbine. The electric power is transmitted to the external grid via individual, isolated air-cooled generator buses incorporating a generator breaker, and a main transformer.

The exhaust from the low pressure turbine cylinders flows to the main turbine condenser which has three shells, located under the exhaust hoods of the low pressure turbine cylinders. The condenser is designed to accept also the steam flow from the main steam bypass system on startup, hot standby and turbine trip. During normal power operation, the steam flow to the condenser amounts to about 60% of the total steam flow, but the condenser system is designed to accommodate the full steam flow for a limited time period; the steam flow shall be reduced to 60% within 20 seconds to avoid a reactor trip due to too high condenser pressure.

The condenser is cooled by the circulating water system which typically incorporates three electrically driven pumps; loss of one pump will call for a power reduction, but will not yield a turbine trip in the short term.

4.3.3.2 Condensate and feedwater systems

The condensate is pumped forward to the deaerator (or the feedwater tank) through low pressure heaters and a condensate clean-up system with ion exchange filters by means of three 50% condensate pumps. The drainage from the heaters is pumped forward through the clean-up system by means of a dedicated low pressure drain pump.

The feedwater system consists of the main feed pumps, two high pressure feedwater heaters, and associated piping. There are three 50% electrical motor driven main feed pumps, drawing from the deaerator (the feedwater tank). Drainage from the high pressure heaters is routed to the deaerator. The power supplies to the FW pumps are utilizing static converter units which eliminate the large inrush currents at direct on line starting and therefore reduces the requirements on "voltage stability" (or rather short circuit strength) of the auxiliary power supply system busbars. Feedwater flow control is achieved by adjusting feed pump speed and the feedwater flow control valves.

Extraction steam for the deaerator and high pressure heaters is provided from high pressure turbine extraction points, including moisture separator drainage and steam reheater exhaust; the low pressure heaters are supplied from extractions on the low pressure turbines.

4.3.3.3 Auxiliary systems

No information has been provided on auxiliary systems, such as the condensate clean-up system, for the turbine plant.

Leakages and drains in the nuclear plant processes are collected in a liquid waste system that is designed to permit maximum reuse of water in a simple process; most of the collected water is chemically pure and may be reused as processed demineralized water after treatment in filters and ion exchangers. Excess water and "unclean" water is discharged, if its "quality" is acceptable, i.e., with low radioactivity, and low content of other unacceptable products; otherwise, it is passed through an evaporator. Then the cleaned water can be reused or discharged; the evaporator residues are conveyed to the solid waste system.

The solid waste system comprises equipment for handling, sorting and compacting low level waste and for solidifying medium level waste originating from the plant, e.g., evaporator residues, ion exchangers and filter aids, always with an ambition of achieving small waste volumes. There is also an offgas system for treatment (delay and filtering) of potentially radioactive gases before releases to the atmosphere.

4.3.4 Instrumentation and control systems

4.3.4.1 Design concept, including control room

Modern process control and communication technology is applied to the BWR 90 - its control and instrumentation systems are mainly based on micro-computers. Communication with the control room or the process systems is realized by means of distributed functional processors. Thus, the protection and control system configuration is characterized by decentralization. The arrangement satisfies the requirements of redundancy and physical separation. It includes intelligent self-monitoring of protective circuits.

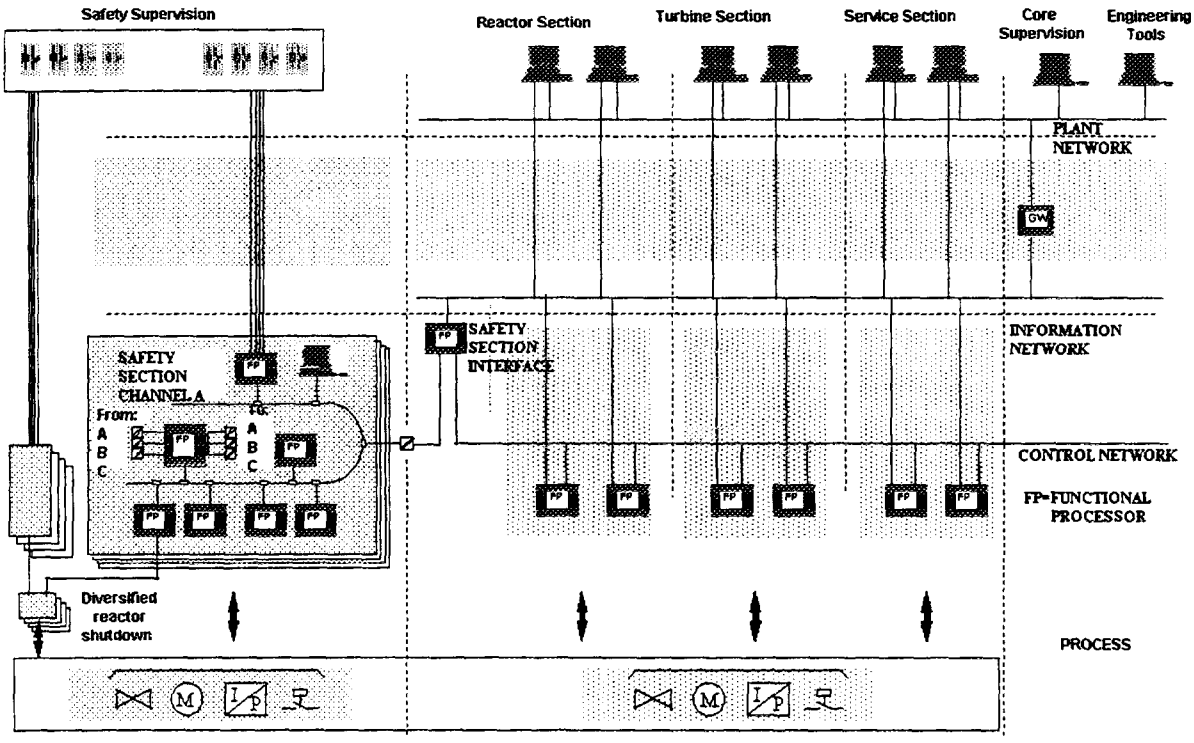


Figure 4.3-3 Plant instrumentation and control system structure

The structure of the instrumentation and control systems of the BWR 90 plant is depicted in Figure 4.3-3.

The use of serial communication links guarantees interference-free performance and reduces cabling. Using hardware modules and basic software from a standard industrial digital system which is available on the market, minimizes maintenance and the necessary stock of spare parts. The arrangement will also tend to improve availability, since components can be replaced quickly and simply by using the engineering tools included in the standard system..

A very important aspect is that the application software is also standardized to simple program functions. This makes it easy even for a non-"computer specialist" to manage the control system design, and it will also simplify implementing new micro-computer generations in the future.

The decentralized configuration, combined with the use of isolation devices, reduces the safety concern of a damaged control room. If the control room should become unavailable, the operating personnel may supervise the process from a separate emergency monitoring centre. The concept allows substantial reduction of space, and has resulted in savings in terms of reduced building volumes.

The man-machine communication in the control room is facilitated by a consistent use of video display units (VDUs), keyboards, and display maps. The main control room contains several work positions, each equipped with a number of VDUs. Typically, one VDU will display a total view of the process in interest, another will provide a list of alarms, and a third VDU will display a diagram with sufficient detail to facilitate operator action.

This arrangement is supplemented with a special overview panel, on which an "overview" of plant functions and status is provided by conventional instruments as well as computer-based VDU displays (VDU projections or EL displays).

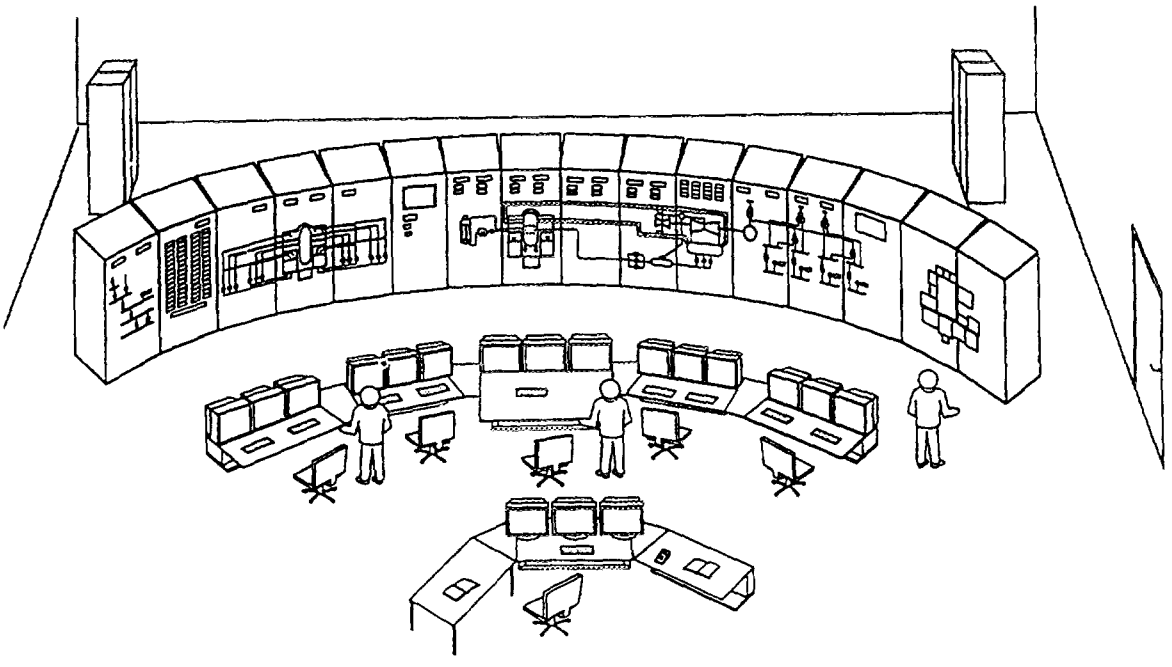


Figure 4.3-4 Main control room arrangement

The overview presentation shows the main process in the form of a flow diagram and indicates the status (normal, disturbed or failed) of various plant functions in correspondence with the operating instructions for the plant. It is visible to all operators in the control room, as indicated on Figure 4.3-4.

The status of safety related systems and functions is presented in a similar way, in accordance with the organization of the Emergency Operation Procedures (EOP). The parameters that are of immediate interest in a disturbance situation, are presented in a direct form. This means that the reactor pressure vessel with in- and outflow connections, together with neutron flux, water level, and reactor pressure, as well as control rods fully in (or not), are displayed directly. Other safety functions are indicated as normal, disturbed or failed in a similar way as for the plant overview, with detailed information at the reactor operator's desk.

In this context it can be noted that the digital RPS (Reactor Protection System) is diversified for events with a frequency of more than once per reactor life time. Diversity can include hard-wired as well as other digital technology.

The main computer has the task of collecting information from the process control systems, and it communicates with the distributed micro-computers via serial links. The main computer compiles information and generates reports, such as daily & weekly operation reports, reports of periodic testing, actual status reports, and disturbance reports. During normal plant operation, the main computer will present occurrences on VDU displays in the control room and in a special "observation room".

A plant wide information network is included which is connected to the information network of the control room. In this way the control room staff and the staff in other plant organisation has the same access to data.

4.3.4.2 Reactor protection and other safety systems

The reactor protection system (RPS) and the other safety related control systems are in the same way as for the BWR 75 built up in a four-division concept; process monitoring, signal treatment and conditioning take place in four independent channels (or divisions). Trip functions are

generally generated in two-out-of-four coincidence logics in each individual "outgoing" division for all RPS functions, except for the emergency core coolant (ECC) injection; the latter circuit operates without the coincidence logic in a "one-out-of-one" mode within each division since an initiation of an ECC division would not significantly impair the operation of the reactor.

The most visible difference compared with the predecessor is the deployment of digitised technology; the control equipment structure is based on utilisation of a range of standardised micro-computer units. As noted above, the basic software of these units are generally standardised to simple program functions in such a way that each process function can be realised in a similar way as with hard-wired equipment; the designer can easily transfer the desired process flow logic to the digitised system. One significant advantage of this approach is that the standardised micro-computer units can be thoroughly tested before delivery, whereas the cabling, or the logic connections, can be checked and verified in an independent operation - in a traditional way. Another advantage of the digitised approach chosen is that the control systems signals are fully "normalised" to low energy signals; the risk of voltage surges in equipment in the control room area is reduced significantly.

The digitisation of the RPS and other safety related control system functions has raised some concern with the safety authorities, and to this end some functions have been supplemented by a diversified back-up. One example of such diversification is the supplementary diversified scram back-up system mentioned above.

4.3.5 Electrical systems

4.3.5.1 Operational power supply systems

The single line diagram of the electrical power systems is shown on Figure 4.3-5.

For the ordinary power distribution, some simplifications have been introduced compared to previous designs. One design constraint in those plants was that the ratio between direct-on-line starting motor loads and available short circuit power on each busbar must be restricted, for voltage stability reasons.

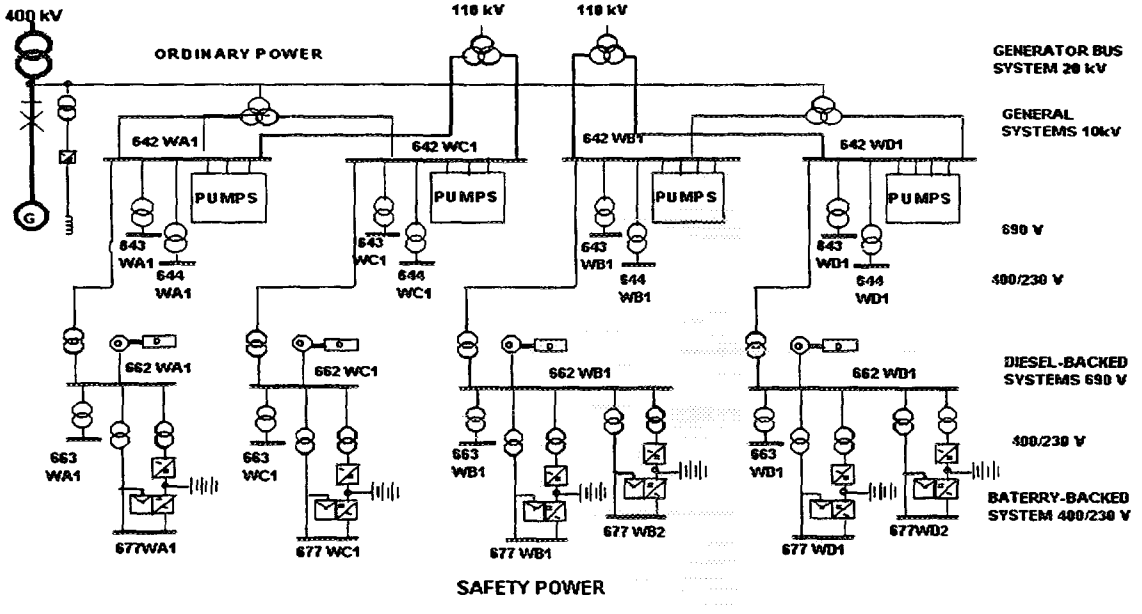


Figure 4.3-5 BWR 90 - Single line diagram

In the BWR 90, the ratings of some of the major plant loads have been reduced by design changes in process systems, and the main feedwater pumps in the turbine plant have been provided with static power supply converters. Modern switchgear components, having higher short circuit current ratings, are also now available, and consequently a significant simplification of the structure of the auxiliary power supply systems has been made possible.

Another visible feature is the simplification at the DC distribution level; DC distributions at several voltage levels for power supply to various types of control equipment have been replaced by power supply from battery-backed AC distribution, using distributed AC/DC converters for the supply to the different types of equipment.

4.3.5.2 Safety related systems

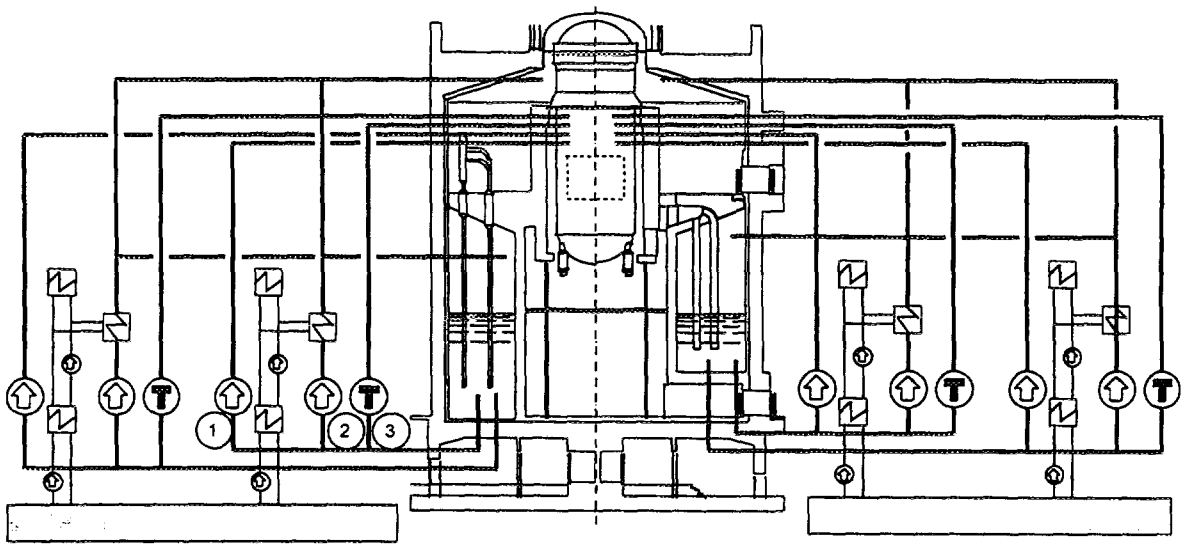
The electrical power systems for safety related objects are strictly divided into four independent and physically separated sub-divisions - a principle that is implemented in the operating BWR 75 plants and maintained in the BWR 90. This means that there are four sets of diesel-backed busbar systems, four diesel generators, four AC/DC-DC/AC converters with intermediate batteries, having sufficient capacity to ensure the power supply to safety related equipment for two hours, and four battery-backed AC busbar systems - all without interconnections.

The structure has been simplified compared with the predecessor; the medium voltage busbar systems have been eliminated and the diesel generators operate at a low voltage level (660 V) following a reduction in size of the pumps of the low pressure coolant injection system, and a number of battery systems and corresponding DC distributions have been replaced by local AC/DC converters supplied with power from the battery-backed AC busbars.

4.3.6 Safety concept

4.3.6.1 Safety requirements and design philosophy

The engineered safety systems in BWR 90 are characterized by their consistent division into four redundant and physically separated subsystems, as illustrated by Figure 4.3-6. This concept that was introduced already in the TVO I and II plants in Finland and further developed in the Forsmark 3 and Oskarshamn 3 plants in Sweden, has been reconfirmed as constituting an optimal arrangement with respect to safety, layout and maintainability.



1: Low Pressure Coolant Injection
2: Containment Vessel Spray System
3: Auxiliary Feedwater System

Figure 4.3-6 BWR 90 - General configuration of the emergency core cooling systems

For the emergency cooling systems, this means that the four subsystems are located in their own bays, adjacent to the reactor containment and surrounded by thick concrete walls. The physical separation is maintained all the way to the ultimate heat sink. The individual compartments for safety related subsystems and components constitute separate fire areas and fire cells. The containment which is of pressure suppression type, is inerted by means of nitrogen during operation.

As in the case of the emergency cooling systems, the safety related auxiliary electrical power supply equipment is divided into four independent and physically separated parts, or subdivisions, and the reactor protection system operates on a 2 out of 4 logic for signal transmission and actuation.

The four safety related, standby power diesel generators with their ancillaries are installed in two diesel buildings, located at opposite sides of the reactor building; this provides a high degree of physical protection with respect to external impacts, eg. against a crashing aircraft. These buildings also house safety related auxiliary power supply and control equipment, as well as pumps and heat exchangers for safety related cooling systems.

The capacities of the emergency core cooling systems suffice to provide water under all postulated pipe break conditions. This statement is also valid assuming that only two of the four redundant subsystems are operable; one is out of operation due to maintenance, etc. and one is postulated to fail (the single failure criterion). The postulated loss-of-coolant conditions include a hypothetical 80 cm^2 leak at the bottom of the reactor vessel. In this context, it can be noted that the capacity of the low pressure coolant injection pumps has been reduced for BWR 90, following comprehensive core cooling analyses. As a secondary effect, it has been possible to simplify the auxiliary power supply systems.

BWR 90 is characterized by diverse means of ensuring the function of safety related systems and components, including, inter alia, the use of diverse types of valves for pressure relief, and the filtered containment venting system for residual heat removal. This contributes to making the plant insensitive to the occurrence of common cause failures. Another example of diversification is the reactor shutdown systems. Shutdown can be achieved in three different ways: through hydraulic scram (which rapidly inserts the control rods), through electro-mechanical insertion (using the motor drives of each rod), and by using the liquid control system (borating the primary loop). The liquid control system is now automatic and has the capability to shut down the reactor during anticipated transients without any movements of the control rods. These diverse means of shutting down the reactor, together with the capability of rapid recirculation flow rate reduction (by pump runback), provide an efficient protection against ATWS (Anticipated Transient Without Scram) events.

In addition to deterministic analyses, and simulations performed with thermal-hydraulic computer codes, a level 1 PSA study has been performed for the BWR 90. It was adapted to the off-site electrical power grid conditions at the Olkiluoto nuclear power station in Finland and to recent data on common cause failures obtained from research work sponsored by the Swedish Nuclear Power Inspectorate (SKI). The PSA study addresses internal events only since previous studies have demonstrated that external events do not contribute significantly to the core damage probability. The PSA shows that LOCA events leading to core melt are extremely unlikely which is typical for BWRs with internal recirculation pumps.

There is a fairly even distribution of contributions from various transients, and the total core damage frequency is significantly below 10^{-5}, in accordance with international guidelines for new reactors (eg. as expressed in INSAG 3). (The core damage frequency has been calculated to $7.4 \cdot 10^{-6}$ per year, using a TVO model for the electrical systems, application of Forsmark 3 site data yielded about $2 \cdot 10^{-6}$, with a decreased contribution from loss of offsite power.)

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

With respect to design for safety, an important point of discussion in the nuclear community in recent years relates to the concepts of redundancy, diversity, and passivity. All the three concepts are associated with pros and cons. Briefly, redundancy and physical separation of safety systems increase reliability, mitigate the effects of external events, and tend to relax the need for quick repair. Extensive separation also minimizes the risk of undesirable system interaction. Full credit from extensive redundancy may be limited by the occurrence of common cause failure (CCF).

Diversifying safety functions is an effective means to avoid sensitivity to CCFs, but since it tends to increase the number of types of equipment, it might be detrimental to simplification and cost reduction. There is also the risk of faulty maintenance and repair. Finally, passivity means the use of systems independent of power sources and operator action. In particular, the issue of human error has appeared to represent a main argument in favour of passive safety functions. While the introduction of passivity in this sense certainly is worth while exploring, as is now done in several design concepts, it should be kept in mind that the operator must not be prevented from responding, and acting, intelligently in emergency situations.

It is necessary to strike a balance among these design aspects and to implement that balance in specific designs. Since BWR 90 is based mainly on technology used in operating plants, the balance here leans towards redundancy, separation and diversity. Inherent safety features such as distinctly negative reactivity coefficients for the reactor core during all operating conditions, utilisation of heat-resistant materials and equipment with low fire risk, and avoiding high fire loads in safety areas of the plant, represent traditional features of nuclear safety engineering, and remain cornerstones for the reactor designs. Passivity is also embodied in a number of design features; the reactor coolant system arrangement provides reliable cooling of the core in natural circulation at a rather high power level, and systems introduced to mitigate the effects of a severe accident (see below) have been designed for passivity.

A high degree of automation has been pursued, as in the BWR 75 design; no operator action is needed within 30 minutes of a disturbance that could threaten safety barriers. With respect to securing the normal operation of the plant, a design goal of some 10 minutes is pursued.

The emergency core cooling systems, the auxiliary feedwater system (the high pressure coolant injection system) and the low pressure coolant injection system, are both active systems, each consisting of four independent and physically separated trains. The first utilises piston type pumps, and the other centrifugal pumps. The containment spray system also comprises four independent and physically separated trains, connected to the gas compression chamber in the wetwell; three of the trains may, on operator action, be re-routed to spraying in the upper drywell. The diesel-backed (safety grade) intermediate closed cooling water systems and service water systems are built in accordance with the same principles; there are four independent and physically separated trains and no interconnections.

Other safety related systems of less importance, such as the standby liquid control system (for injection of boron solution into the moderator), are laid out in a less comprehensive manner, with two separated trains.

4.3.6.3 Severe accidents (Beyond design basis accidents)

When the design review of the BWR 90 was initiated, regulatory developments indicated a need to strengthen the capability of the reactor containment to withstand the effects of a core melt accident. Such requirements are now codified in a number of European countries, among these Finland, France, Germany and Sweden.

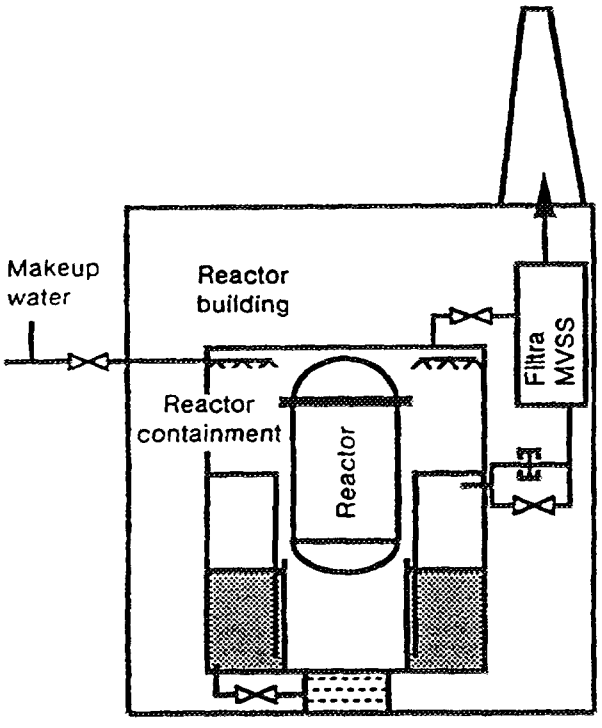


Figure 4.3-7 Severe accident mitigation features

The essential features of the BWR 90 containment (Figure 4.3-7) that have been introduced to achieve an enhanced containment capability to confine radioactive products and protect the environment against large releases of radioactive matter, also in the event of a degraded core accident, are:

1. The blow-down of steam to the suppression pool passes through vertical concrete pathways to horizontal openings between drywell and wetwell.
2. The relief pipes from the safety/relief valves are drawn into the suppression pool via the lower drywell rather than penetrating the drywell-wetwell intermediate floor.
3. A pool is provided at the bottom section of the lower drywell for the purpose of collecting and confining fuel melt debris. The pool is provisionally assumed to be permanently filled with water to enhance passive safety.

These arrangements improve the reliability of the pressure-suppression system and reduce the probability of containment leakage during an accident. In addition, the containment vessel can be vented, manually or passively through a rupture disc, to the stack through a filter system, installed in the reactor building. This filter is similar to the filtered venting systems installed at all nuclear power plants in Sweden. Arrangements are also made to enable filling the containment with water to the level of the top of the core, in order to establish a final stable state following a severe accident involving core damage; this water is supplied to the containment spray system and the providing system uses a completely independent water source and power supply.

4.3.7 Plant layout

4.3.7.1 Buildings and structures, including plot plan

The plant and buildings of the BWR 90 are laid out and designed to satisfy aspects of safety, maintenance and communication in a balanced way. The layout is strongly influenced by safety requirements, in particular the physical separation of safety related equipment.

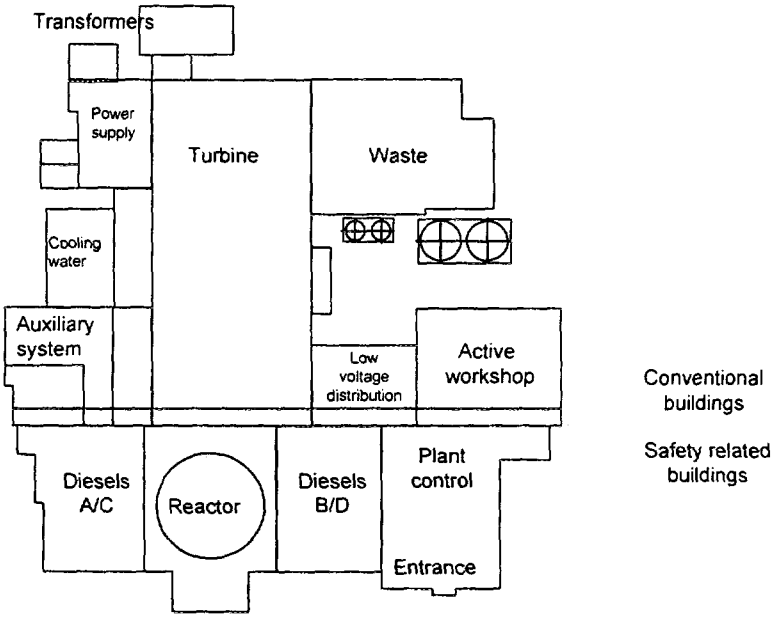


Figure 4.3-8 General layout, plot plan

With respect to building layout and arrangement ABB Atom has traditionally favoured a co-ordinated and compact building complex; the number of doors and transport openings, release points, transport routes etc. can be kept low and supervision becomes easier.

The general arrangement of the buildings is depicted in Figure 4.3-8. It is characterized by a division into an essentially nuclear and safety related portion, consisiting of the reactor building, the diesel buildings and the control building, and a more conventional portion that comprises the turbo-generator and auxiliary systems of the plant. The "conventional" part is separated from the former by a wide communication area. This arrangement is advantageous when building the plant as well as during plant operation, since the conventional part does not interfere with the nuclear part.

The building arrangement is also characterized by a system of communication routes connected to the wide communication area for personnel and equipment, between and inside buildings, that serves to facilitate maintenance, inspection and repair work by ensuring good accessibility to plant equipment.

Together with a suitable design and installation of the process systems, a meticulous choice of materials, a proper routing of ventilation air flows, this paves the ground for achieving low operational radiation exposure. The BWR 75 plants, the forerunner to the BWR 90, have been operated at an annual occupational exposure of 1 mansievert or lower, and the BWR 90 design aims at lowering it even further; the ambitious target is an average annual exposure of only 0.5 mansievert.

The safety related portions of the building complex, the reactor building with the reactor containment, the adjacent diesel buildings, and the control building, are designed to withstand the effects of earthquakes. The standard nuclear island is designed to sustain a "safe shutdown earthquake" (SSE) of 0.25 g; higher SSE levels can be accommodated by further strengthening of structures and some design modifications.

The design strength of the nuclear island structures varies with the site conditions; seismic "activities" may require additional amounts of reinforcement, and protection against a crashing aircraft would call for a strengthening of outer structures. With respect to the reactor the aircraft should as far as possible be prevented from penetrating the first line of defence - the walls or the roof of the reactor building, while a combination of separation by distance and protection by strong walls and roofs will be the preferred strategy for other safety related equipment.

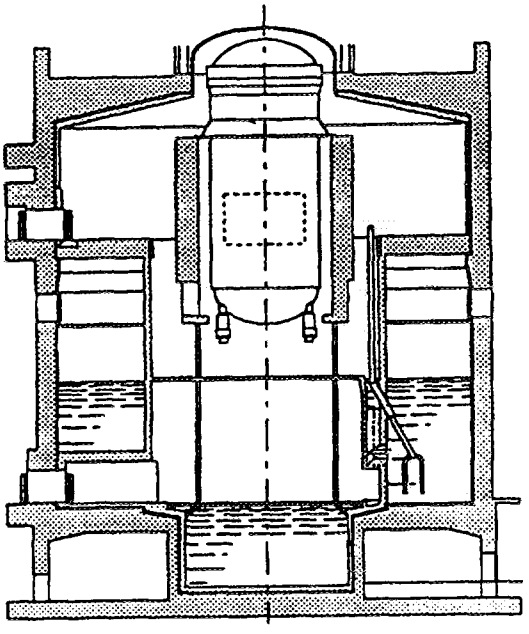


Figure 4.3-9 Containment

4.3.7.2 Reactor building

The reactor building encloses the primary containment with the primary circuit and houses all primary process and service systems for the reactor, such as handling equipment for fuel and main components, fuel pools, reactor water cleanup system and engineered safety systems.

In comparison with previous plants, a substantial reduction of building volumes has been achieved, implying a significant cost reduction. Nevertheless, BWR 90, like previous plants, is characterized by a fairly spacious layout, which ensures easy access to the plant components. The installation and ventilation principles are maintained and the material specifications even more stringent; hence, low occupational exposures are anticipated also for the BWR 90 plants.

As noted above, the reactor building encloses the primary containment completely and is designed to serve as a secondary containment, kept at underpressure by ventilation systems in which the exhaust air can be rerouted to filters when needed. The top of the reactor building serves as a reactor service room with pools for reactor service, for storage of internals during refuelling, and for storage of spent fuel and "failed" irradiated components, space for tools and handling equipment and cranes and platforms (refuelling and service bridges, overhead crane). A receiving station and storage for fresh fuel is arranged at a lower level of the reactor building, with a lifting shaft to the reactor service room.

4.3.7.3 Containment

The primary system, the reactor coolant pressure boundary, and important ancillary systems are enclosed in the primary containment, a cylindrical prestressed concrete structure with an embedded steel liner - as in all previous ABB Atom plants. The containment vessel, including the pressure-suppression system and other internal structural parts as well as the pools above the containment, forms a monolithic unit (Figure 4.3-9) and is statically free from the surrounding reactor building, except for the common foundation slab. A steel dome provides a "removable" closure of the shaft above the reactor pressure vessel.

The primary containment is of pressure-suppresion type, with two major compartments - a drywell and a wetwell. The drywell represents the volume that surrounds the RPV, with an upper

portion (basically, extending from a level flush with the bottom of the core and upwards) and a lower portion located below the RPV (and below the core). The wetwell is separated from the drywell by a partition floor and a cylindrical wall; the lower portion of this separated volume is filled with water - the condensation pool, whereas the upper portion serves as a gas compression chamber. In the event of drywell pressurization, eg. due to a LOCA inside the containment, drywell atmosphere together with steam will be pushed into the condensation pool via a horizontal passage arrangement through the separating wall; non-condensibles will collect in the gas compression chamber whereas the steam will condense in the pool water.

The blowdown pipes from the safety (relief) valves in the pressure relief system are routed through these horizontal passages, leaving the partition floor without penetrations; the probability of a degraded pressure suppression function has been reduced to a very low level.

The pressure suppression function is supported by the four train containment spray system that is continuously in service, with one train supplying spray water from the condensation pool to the gas compression chamber; in accident situations the system will start operation at full capacity. Spraying is also possible for the upper drywell - after rerouting, on operator action. The drywell spray is generally initiated only in the event of "small" LOCAs to "depressurize" the containment.

Equipment containing reactor water at high pressure and high temperature is located inside the containment, which is designed to withstand the pressures and temperatures that may occur after a rupture of the largest pipe; the reinforced structure is quite strong and can withstand also impacts of a crashing aircraft.

The containment design of BWR 90 incorporates also some features that aim at protection of the public and the environment against major releases of radioactive material even in severe accident situations involving core degradation and core damages. To this end the containment has been provided with an overpressure protection system which by means of rupture disks automatically, and in an entirely passive way, will relieve excessive pressure to the stack via a filter system; this will prevent serious land contamination also in such very unlikely situations. Besides, the central, lowermost portion of the lower drywell has been made as a pool (with or without water during operation) with cooled surfaces; this volume serves to collect, confine and cool possible molten debris from the reactor in such accident situations. In this context, it can be noted that cooling water for this purpose can be provided by gravity drain from the condensation pool or the reactor service pool on top of the containment, and additional makeup water can be provided from outside after things have become more quiet.

4.3.7.4 Turbine building

The turbine building of the BWR 90 is directly adjacent to the reactor building, and the main turbine generator shaft points away from the reactor proper. The HP turbine is located at the reactor building end, and the generator at the other; isolated phase buses lead from the generator to the generator circuit breaker and from there to the main (stepup) transformer, with branching offs to two three-winding plant transformers for the auxiliary power supply. The main transformer and the plant transformers are located immediately outside the turbine building.

The building itself represents a rather conventional structure; there are no safety related requirements. The different auxiliary systems are installed at various locations and levels; e.g., the condenser and the condensate pumps are at the very bottom of the building, and the feedwater tank at a high location, the feedwater pumps are at ground level and the high pressure feed heaters rise from below grade towards the turbine hall floor. In addition to the turbine systems, the building contains also, below grade and at the far end from the reactor, the equipment of the condensate clean-up system and the offgas system. The pumps for the circulating water, for the service water and closed cooling systems are installed in a pump building immediately outside the turbine building.

4.3.7.5 Other buildings

Diesel buildings

The diesel buildings are located on two sides of the reactor building and structurally integrated with it to enhance the protection against the effects of earthquakes. The two diesel buildings contain most of the equipment of the safety related systems that is located outside the reactor building; they house the four divisions of diesel generators and busbars of the diesel-backed auxiliary power supply system, the AC/DC-DC/AC converters, batteries and busbars of the battery-backed power supply system, the safety related control equipment, pumps, valves and heat exchangers for the intermediate closed cooling systems, and pumps and valves for the service water system. The buildings basically consist of two parts, each containing one safety division. The innermost part (as seen from the front of the reactor building complex) is provided with a strengthened structure to protect against a crashing aircraft; this strengthening and the location on two sides of the reactor building will prevent a crashing aircraft from damaging more than one division.

Control building

The control building which is located at one side of the reactor building, with one of the diesel buildings in between, houses the main control room and its ancillaries, the centre for handling work permits for maintenance activities in the plant, and the entrance controls for access to the controlled and uncontrolled areas of the plant. It is the main entrance to the plant building complex and contains a reception desk and dressing rooms for personnel with lockers and showers etc. As an optional feature a visitors gallery with windows overlooking the control room area may also be provided.

The control building is not safety classified per se, but steps have been taken to ensure that the control room, with its equipment and in particular its operators, is well protected against external impacts such as earthquakes and crashing airplanes, and also against possible detrimental gases in the ambient air.

Waste building

The waste building is located adjacent to the main building complex. It contains the systems and equipment for receiving and treating liquid and solid low and medium level radioactive waste from the plant. Demineralised water (process water) is as far as possible cleaned (in filters and ion exchangers and routed back to the plant for re-use; too contaminated water is routed to evaporators for treatment. Water from laundries etc. is cleaned and released to the cooling water outlet channel, together with excessive cleaned process water. Evaporator residues and ion exchangers and filter aids are transferred to a solid waste system for solidification in concrete and subsequent on-site or off-site storage.

Other auxiliary buildings

The plant encompasses a number of other buildings for auxiliary systems and equipment: a medium voltage switchgear building (with the four MV busbar systems of the operational power supply system and associated control and protection devices); a low voltage switchgear building (with the low voltage busbar systems of the operational power supply system and associated control and protection devices); a cooling system building (with the pumps, valves and heat exchangers of the non-safety service water and closed cooling water systems); a cooling water intake building (with screens for the water intake); a workshop building (with workshops for active and non-active components); and a storage building (for spare parts and consumables).

4.3.8 Technical data

General plant data		
Power plant output, gross	1 238	MWe
Power plant output, net	1 197	MWe
Reactor thermal output	3 300	MWt
Power plant efficiency, net	36.3	%
Cooling water temperature	5	°C
Nuclear steam supply system		
Number of coolant loops	1	
Primary circuit volume	~480	m^3
Steam flow rate at nominal conditions	1 780	kg/s
Feedwater flow rate at nominal conditions	1 780	kg/s
Reactor coolant system		
Primary coolant flow rate	13 100	kg/s
Reactor operating pressure	7.0	MPa
Steam temperature/pressure	285/6.9	°C/MPa
Feedwater temperature	215	°C
Core coolant inlet temperature	278	°C
Core coolant outlet temperature	286	°C
Mean temperature rise across core	8	°C
Reactor core		
Active core height	3.75	m
Equivalent core diameter	4.60	m
Heat transfer surface in the core	7 676.8	m^2
Average linear heat rate	12.57	kW/m
Fuel weight	126	t U
Average fuel power density	26.2	kW/kg U
Average core power density	52.5	kW/l
Thermal heat flux, F_q	429.9	kW/m^2
Enthalpy rise, F_H	-	

Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	3.99	mm
Rod array	4 × (5×5) in square lattice	
Number of fuel assemblies	700	
Number of fuel rods/assembly	4×5×5	
Number of spacers	4×6	
Enrichment (range) of first core, average	3.35	Wt%
Enrichment of reload fuel at equilibrium core	3.1	Wt%
Operating cycle length (fuel cycle length)	12	months
Average discharge burnup of fuel	50 000	MWd/t
Cladding tube material	annealed, recrystallised Zr 2	
Cladding tube wall thickness	0.63	mm
Outer diameter of fuel rods	9.62	mm
Fuel channel/box; material	Zr-4	
Overall weight of assembly, including box	~294.8	kg
Uranium weight/assembly	180.0	kg
Active length of fuel rods	3.75	mm
Burnable absorber, strategy/material	axial and radial grading/ Gd_2O_3 mixed with fuel	
Number of control rods	169	
Absorber material	B_4C/Hafnium	
Drive mechanism	electro-mechanical	
Positioning rate	30	mm/s
Soluble neutron absorber	Boron	
Reactor pressure vessel		
Inner diameter of cylindrical shell	6 400	mm
Wall thickness of cylindrical shell	135 + 5	mm
Total height, inside	20 900	mm
Base material: cylindrical shell	low-alloy carbon steel	
RPV head	[to ASTM A533, grade B, ASTM A508, class 3, or equiv.]	
lining	stainless steel	
Design pressure/temperature	8.5/300	MPa/°C
Transport weight (lower part)	655	t
RPV head	130	t

Reactor recirculation pump

Type	Integral, glandless, centrifugal	
pump		
Number	8	
Design pressure/temperature	8.5/300	MPa/°C
Design mass flow rate (at operating conditions)	1 650	kg/s
Pump head	*0.2*	MPa
Rated power of pump motor (nominal flow rate)	840	kW
Pump casing material	as for RPV	
Pump speed (at rated conditions)	1 480	rpm
Pump inertia		kg m^2

Primary containment

Type	Pressure-suppression	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	25.5/41.2	m
Design pressure/temperature	600/172	kPa/°C
Design leakage rate	0.5	vol%/day
Is secondary containment provided?	yes	

Reactor auxiliary systems

Reactor water cleanup,	capacity	2 x 35	kg/s
	filter type	deep-bed	
Residual heat removal,	at high pressure	2x55	kg/s
	at low pressure (100 °C)	2 x 22	MW
Coolant injection,	at high pressure	4 x 22.5	kg/s
	at low pressure	4 x 240	kg/s

Power supply systems

Main transformer,	rated voltage	24/400	kV
	rated capacity	3 x 500	MVA
Plant transformers,	rated voltage	24/10/10	kV
	rated capacity	2 x 40/20/20	MVA
Start-up transformer	rated voltage	110/10/10	kV
	rated capacity	1 x 40/20/20	MVA
Medium voltage busbars (6 kV or 10 kV)		4	
Number of low voltage busbar systems		8	
Standby diesel generating units: number		4	
rated power		2 800	MW
Number of diesel-backed busbar systems		4	
Voltage level of these		690	V AC
Number of DC distributions		0	
Voltage level of these			V DC
Number of battery-backed busbar systems		4	
Voltage level of these		3 × 230	V AC

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	axial, double flow, condensing	
Number of turbine sections per unit (e.g. HP/LP/LP)	HP/3 LP	
Turbine speed	1 500	rpm
Overall length of turbine unit	~35	m
Overall width of turbine unit	~ 6	m
HP inlet pressure/temperature	6.71/282	MPa/°C

Generator

Type	3-phase, turbo-generator	
Rated power	1 383	MVA
Active power	1 238	MW
Voltage	24	kV
Frequency	50	Hz
Total generator mass, including exciter	859	t
Overall length of generator		m

Condenser

Type	shell type	
Number of tubes		
Heat transfer area	45 120	m^2
Cooling water flow rate	48 000	m^3/s
Cooling water temperature	5	°C
Condenser pressure	3.2	kPa

Condensate pumps

Number	3	
Flow rate	~540	kg/s
Pump head	~1.4	MPa
Temperature	~30	°C

Pump speed		rpm

Condensate clean-up system

Full flow/part flow	full condensate flow	
Filter type	rod type	

Feedwater tank

Volume	300	m^3
Pressure/temperature	4.0/130	MPa/ °C

Feedwater pumps

Number	3 × 50%	
Flow rate	890	kg/s
Pump head	~6.5	MPa
Feed pump power	9.4	MW
Feedwater temperature (final)	215	°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages,	low pressure	3 × 2
	high pressure	2 × 2
	feedwater tank	1

4.3.9 Project status and planned schedule

The development of the BWR 90 started in 1986 as a review of the "lessons learned" from previous plant projects; in particular, from designing and commissioning the Forsmark 3 and Oskarshamn 3 nuclear power plants in Sweden. The conceptual design, and most of the basic design has been completed, and it was offered to Finland, as one of the contenders for the fifth Finnish nuclear power plant project. Since 1988, the development work was conducted in co-operation with the Finnish utility TVO (Teollisuuden Voima Oy) which is operating two ABB Atom BWR plants - the TVO I and II units. These units are two of the top performing nuclear power plants in the world with an average capacity factor of more than 90% over the last 9 years; this has ensured an efficient feedback of operating experience.

As noted, the BWR 90 is based on the design, construction, commissioning and operation of a number of BWR plants in Finland and Sweden, and it has been developed by making specific changes to an established reference design, the Forsmark 3 and Oskarshamn 3 power plants. The experience from these plants was reviewed thoroughly with the aim of evaluating possible improvements, simplifications and cost reductions. The resulting plant has significantly reduced building volumes, shortened construction time, and decreased amounts of systems and components. Measures for simplified operation, testing and maintenance have also been included; therefore, the modified design offers lowered costs and more simple operation, compared with the earlier design. Still, modifications to the BWR 75 plant design are mostly moderate, and therefore, the operating records of the BWR 75 plants can be drawn upon in the licensing of BWR 90 plants.

A precise classification of the project status is not easily done; the status of the BWR 90 project lies somewhere between completed basic design and completed detailed design. To be more specific, construction and commissioning QA programmes have been developed, and the plant design is complete, except for some minor items, but detailed design/engineering has not been made for all systems and components, and detailed design, or detailed specifications, for procurement of all materials, components, systems, package units, construction/erection services, etc. have not yet been completed. Current design and engineering activites are being focussed on detailed analyses of severe accident sequences, studies of the possibilities for further improvement of the containment, procedures for the verification of digitised control equipment, and optimisation of plant construction activities.

With respect to licensing, it may be noted that the authorities in Sweden never take on licensibility reviews, until a utility files for a construction permit, and a design certification process does not exist. Reference is therefore made to the comment above on the close relationship with the BWR 75 design, and to the licensing discussions that have taken place with STUK, the Finnish licensing authority, in connection with the "Finland V" project.

Construction of a BWR 90 plant can build directly on the experience gained from previous projects. The construction activities have been analyzed by the team of civil engineering, installation and commissioning supervisory personnel that built and commissioned the Oskarshamn 3 Nuclear Power Plant in Sweden in 57 months from the first pouring of concrete to start commercial operation.

The resulting schedule for this BWR 90 plant version shows that the total construction time for the plant - from pouring of the first structural concrete to start commercial operation - will be 48 + 6 months, i.e., the total construction time stipulated in the EPRI URD will be met.

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4.4 EP 1000, WESTINGHOUSE, USA/GENESI, ITALY/EUR

4.4.1 Introduction

In 1994, a group of European utilities initiated, together with Westinghouse and its industrial partner GENESI (an Italian consortium including ANSALDO and FIAT), a programme designated EPP (European Passive Plant) to evaluate Westinghouse passive nuclear plant technology for application in Europe. The European utility group consists of the following organizations:

- Agrupación Eléctrica para el Desarrollo Tecnologico Nuclear (DTN), Spain
- Electricité de France, France
- ENEL, SpA., Italy
- Imatran Voima Oy, Finland
- Scottish Nuclear Limited (acting for itself and on behalf of Nuclear Electric plc), U.K.
- Tractebel Energy Engineering, Belgium
- UAK (Represented by NOK-Beznau), Switzerland
- Vattenfall AB, Ringhals, Sweden

The european passive plant programme involves evaluation of the Westinghouse 600 MWe AP-600 and 1000 MWe simplified pressurized water reactor (SPWR) designs against the European Utility Requirements (EUR), and when necessary, to investigate possible modifications to achieve compliance with the EUR. In Phase 1 of the programme which has been completed, the following major tasks were accomplished:

1) The impacts of the European Utility Requirements (EUR) on the Westinghouse nuclear island design were evaluated.
2) A 1000 MWe passive plant reference design (EP 1000) was established which conforms to the EUR and is expected to be licensable in Europe.

With respect to safety systems and containment, the reference plant design closely follows that of the Westinghouse SPWR design, while the AP-600 plant design has been taken as the basis for the EP 1000 reference design in the auxiliary system design areas. However, the EP 1000 design includes also features required to meet the EUR, as well as key European licensing requirements.

Extensive design and analysis efforts have been completed for the AP-600 and SPWR passive plant designs during multi-year programmes. The results from these programmes will be utilized wherever possible to minimize the additional work to be performed on the EP 1000 design.

The ultimate objective of phase 2 of the programme is to develop design details and perform supporting analyses to produce a standard safety analysis report (SSAR) for submittal to European Safety Authorities. The first part of Phase 2 will focus on improving the design of important systems and structures.

Incorporation of the EUR has been a key design goal for the EP 1000 from the beginning of the programme, and it will continue to be so during and beyond the development of the Phase 2. The EP 1000 has a well defined design basis that is confirmed through engineering analyses and testing and is in conformance with the EUR. Some of the high level design characteristics of the plant are:

- The standard design will be applicable to European sites.
- The net electrical power is about 1000 MWe and the thermal power 2910 MWt.
- The average coolant temperature at the reactor vessel exit does not exceed 617°F (325°C) during normal operation even with 10% of the steam generator tubes plugged.
- The reactor core is a low power density core that uses the Westinghouse 12 ft (3658 mm), 17×17 assembly.
- Short lead time and construction schedule are expected.

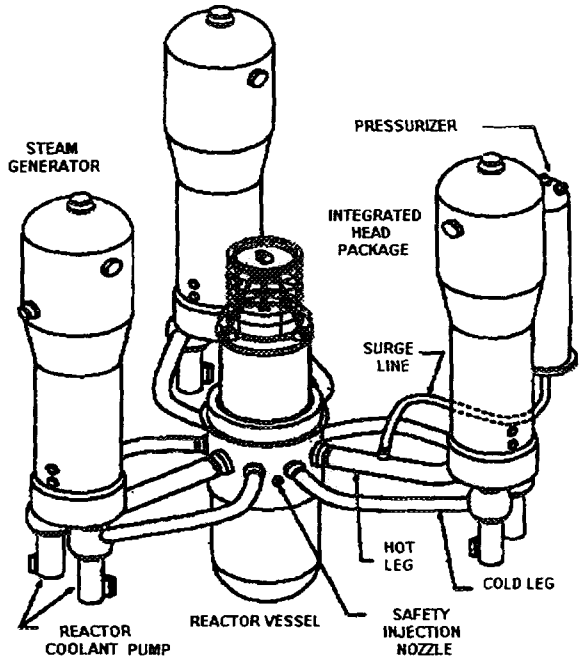


Figure 4.4-1 EP 1000 - Reactor coolant system layout

- No plant prototype is needed since proven power generating system components are used.
- Major safety systems are passive; they require no operator action for more than 72 hours after an accident, and maintain core and containment cooling for a protracted period of time without AC power.
- Predicted core damage frequency is below 10E-5/yr, and frequency of significant release is below 10E-6/yr.
- Occupational radiation exposure is expected to be below 0.7 man Sv/yr (70 man rem/yr)
- The core is designed for a 18 month fuel cycle assuming 85% capacity factor (15.3 EFPM). The core shall be capable of operating with a 24 months fuel cycle.
- A preliminary 24 month Low boron core and a conceptual 18 month 50% MOX core have been designed to cope with the Low boron capability and MOX core requirements of EUR Rev.B
- Refuelling outages without major problems or major maintenance will be conducted in 17 days or less.
- Plant design aims at a lifetime of 60 years without replacement of the reactor vessel
- The design aims at an overall plant reliability greater than 90%, including forced and planned outages; the goal for unplanned reactor trips is less than one per year
- The design incorporates the US advanced light water reactor (ALWR) utility requirements and satisfies the US general design criteria of 10 CFR 50, Appendix A.

4.4.2 Description of the nuclear systems

4.4.2.1 Primary circuit and its main characteristic

The primary circuit of the EP 1000 reactor has retained most of the general design features of current-day designs, but some evolutionary features to enhance reliability, simplicity of operation, maintainability and plant safety have been adapted in the design. An isometric view of the main loop is shown in Figure 4.4-1.

The coolant loops consist of three hot leg and six cold leg pipes and the reactor coolant pumps (RCPs) are mounted directly on the channel head of each steam generator (SG). This allows the pumps and steam generator to use the same structural support, greatly simplifying the support system and providing

more space for pump and steam generator maintenance. The combined vertical support is a single pinned column extending from the cell floor to the bottom of the channel head.

The integration of the pump suction into the bottom of the steam generator channel head eliminates the crossover leg of coolant loop piping, thus avoiding the potential for core uncovery due to loop seal venting after a small loss of coolant accident (LOCA).

4.4.2.2 Reactor core and fuel design

The core, reactor vessel, and reactor internals of the EP 1000, shown in Figure 4.4-2, are similar to those of currently operating Westinghouse PWR plants, but several new features are incorporated to enhance the performance characteristics as compared with existing plants.

The reactor core uses the Westinghouse 12 foot (3658 mm), 17×17 fuel assembly. A low-power density is achieved by making the core larger than previous 1000 MWe designs, with the number of fuel assemblies increased from 157 to 193. This configuration results in core power density and average linear power density enhancements of about 25 percent, to 88.84 kW/l and 15.16 kW/m, over existing plants of the same power rating. This results in lower fuel enrichments, less reliance on burnable absorbers, and longer achievable operating cycles.

The core is surrounded by a stainless steel radial neutron reflector which contributes to lowering fuel cycle cost and extending reactor life. It reduces neutron leakage, thereby improving core neutron utilization and allowing for reduced fuel enrichment and has the added benefit of reducing the neutron fluence on the reactor vessel wall, an important factor in view of the 60 year lifetime design objective.

The current core design allows 18 month refuelling cycles with an 85 percent capacity factor (approximately 466 effective full-power days per cycle) to be achieved and requires no burnable absorbers other than for the first cycle of operation.

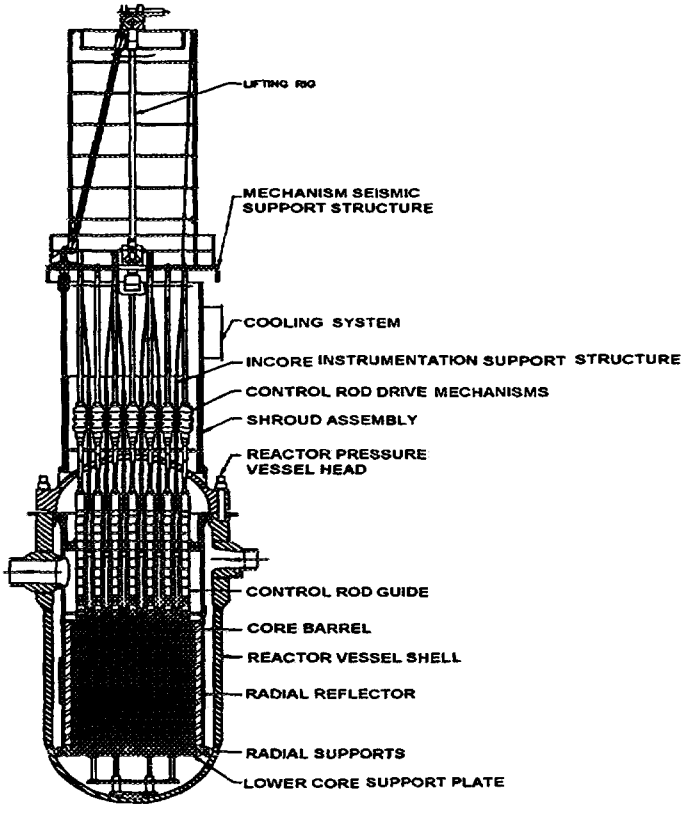


Figure 4.4-2 EP 1000 reactor system

Another core design feature is the use of reduced-worth control rods (termed "gray" rods) to achieve daily load-follow capability without requiring daily changes in the soluble boron concentration. The use of gray rods, in conjunction with an automated load-follow control strategy, eliminates the need for processing thousands of liters of water per day to change the soluble boron concentration sufficiently to achieve a daily load-follow schedule. Except for the neutron absorber materials used, the design of the gray rod assembly is identical to that of a normal control rod assembly.

The control rod drive mechanisms (CRDMs) are the same as used in previous Westinghouse plants. The CRDMs are housed within an integrated head package (IHP), which combines several separate components in one structure. As shown in Figure 4.4-2, the IHP is mounted directly on the reactor vessel head. Use of the IHP and the resulting elimination of several operations contributes to an overall reduction in critical path time during refuelling outages. The design of the IHP is similar to those operating in current plants.

In addition to the reference core configuration, conceptual designs for both Low soluble boron and 50% MOX cores to meet the EUR fuel and core performance criteria have been developed during the Phase 1 of the EPP Program.

4.4.2.3 Fuel handling and transfer systems

Refuelling of the reactor is performed in the same way as for current-day plants. After removing the vessel head, fuel handling takes place from above, using the refuelling machine which moves fuel between the fuel transfer system and the reactor core. The refuelling machine consists of a rectilinear bridge and a trolley crane with a vertical mast extending down into the refuelling cavity. The bridge spans the refuelling cavity and runs on rails set onto the edge of it. The bridge and the mast are moved to position the vertical mast over the fuel assembly to be moved. In addition, the refuelling machine is equipped with an auxiliary hoist which provides additional capability for other refuelling operations. Nuclear fuel assemblies are transferred between the refuelling cavity in the containment building and the fuel transfer canal to the spent fuel pit located in the fuel handling area of the auxiliary building. During refuelling, approximately one third of the core inventory is replaced.

4.4.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel is designed to provide the smallest and most economical volume required to contain the core and the necessay support and flow-directing internals. The vessel is cylindrical, with a welded hemisherical bottom head and a removable flanged-and-gasketed hemisherical upper head. It houses the core, core support structures, control rods and other parts directly associated with the core. The control rods are operated by sealed drive mechanisms mounted on the upper head. The head supports the CRDM drives, the In-core instrumentation system (IIS) guide tubes and the IHP structure.

The vessel has six 22 in. (559 mm) ID inlet and three 31 in. (787 mm) ID outlet nozzles located in two horizontal planes between the upper head flange and the top of the core. The nozzles are located in those planes to provide an acceptable crossflow velocity in the vessel outlet region and to facilitate an optimum layout of the RCS equipment. The inlet and outlet nozzles are offset with the inlet position above the outlet to allow removal of a reactor coolant pump without requiring removal of the core. Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel annulus, turns at the bottom and flows up through the core to the outlet nozzles. The vessel is made of low alloy steel with weld deposited austenitic stainless steel cladding on all surfaces exposed to the reactor coolant.

The reactor vessel closure flange is sealed by two metallic O-rings. Seal leakage is detected by means of two leakoff connections, one between the inner and outer O-rings and one outside the outer O-ring. Piping and associated valves are provided to direct any leakage to the reactor coolant drain tank. Excessive leakage is indicated by a high temperature alarm from a detector in the leakoff line. A local

connection is provided for detection and measurement of small leaks that would not be indicated by the high temperatures.

Reactor internals

The reactor internals are the structural assemblies that support the core within the reactor vessel and provide the proper flow path for the circulation of the coolant through the core. Included with the internals are those structures that guide and enable movement of the control rods.

The internals consist of two major assemblies: the lower internals and the upper internals. The upper internals and the core barrel are supported by the reactor vessel ledge and are restrained against upward movement by the vessel head. The lower end of the internals is restrained against horizontal movement by barrel radial support keys located at the bottom of the core barrel.

The lower internals consists of the core barrel, lower support plate, vortex suppression plate, radial reflector, radial supports and the related attachment hardware. During reactor operation the barrel serves to direct the coolant flow from the reactor vessel inlet nozzle through the downcomer annulus, and into the lower plenum below the lower support plate. The flow then turns and passes upward through the lower support plate into the core region.

The upper internals assembly which is located above the core, consists of the upper support plate, upper support columns, upper core plate,rod cluster control guide tubes and related attachment hardware. During operation coolant flows up from the core through the upper core plate and through the outlet nozzles.

Steam generator.

The steam generator, called Delta 75, is a vertical shell and U-tube evaporator with integral moisture separating equipment based on standard Westinghouse Model F technology.

The steam generators are designed to meet the performance requirements specified for the EPP plant. More specifically, the steam generators shall have sufficient heat transfer surface area to produce a steam pressure at full power of 910 psia (6.28 MPa). Furthermore, the steam generator, at a Best estimate flow (BEF) rate of 99 900 gpm (22,687 m^3/h) of reactor coolant and a feedwater temperature of 430°F (221°C), must produce a steam flow rate of $4.2x10^6$ lb/h (1910 t/h) with a maximum moisture carry-over of 0.25% by weight. The primary coolant enters the SG tubes at about 322 C and it is cooled down to 286 C.

The primary side head has been modified to allow for the direct attachment of two reactor coolant pumps. The two coolant pumps and the SG make one structure providing safety, thermohydraulic and structural advantages; reduction of small break LOCA, reduction of head pressure drop and simplified support structure.

Maintenance and service aspects have also been considered in the SG design. Principal among these are access, radiation exposure, and tooling compatibility associated with operation such as eddy current testing, sludge lancing, in-service inspection, pump maintenance, sleeving, plugging and steam generator replacement.

The steam generator is equipped with a number of state of the art features to enhance steam generator performance and reliability. Among these is the use of inconel 690 thermally treated tubing material. Inconel 690 TT has excellent overall corrosion resistance, primary stress corrosion cracking resistance, and low primary release rates which are expected to reduce primary side radiation levels by 50% compared to units employing Inconel 600 tube material. Another feature is hydraulically expanded tube-to-tubesheet joints, ensuring low stresses and tight joints with controlled crevice depth that minimize the potential for crevice corrosion. All the model F-type steam generators have operated on all volatile treatment secondary side water chemistry. A cutaway drawing of the EP 1000 steam generator is shown in Figure 4.4-3 .

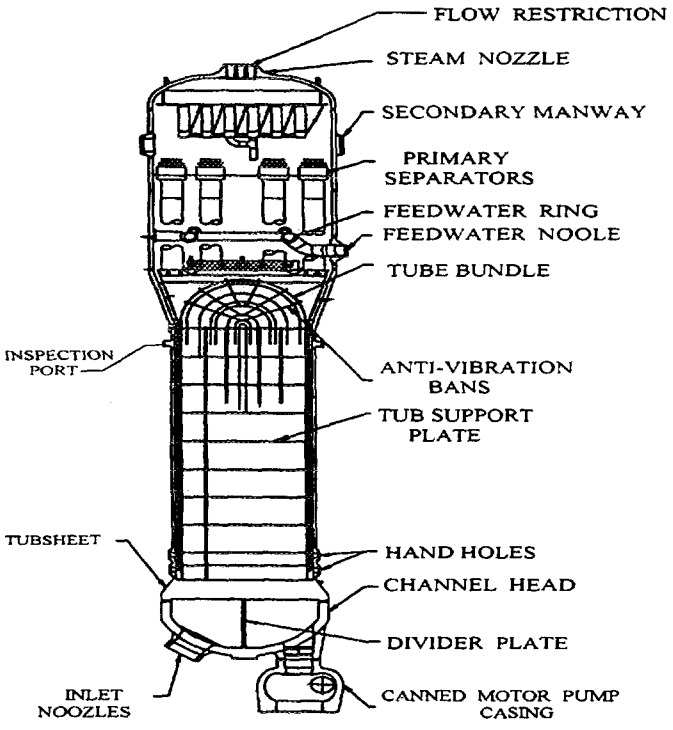


Figure 4.4-3 Improved EP 1000 steam generator, based on proven Model F design

Pressurizer.

The pressurizer is of conventional design, based on proven technology and years of operating experience. The pressurizer is about 30 percent larger than that normally used in a plant of comparable power rating. The larger pressurizer increases transient operation margins, resulting in a more reliable plant with fewer reactor trips, avoiding challenges to the plant and operator during transients. It also eliminates the need for fast-acting power-operated relief valves, which are a possible source of RCS leakage and maintenance.

Reactor coolant pumps

The reactor coolant pump (Figure 4.4-4) is a canned motor design incorporating the latest commercial and marine canned motor pump technology.

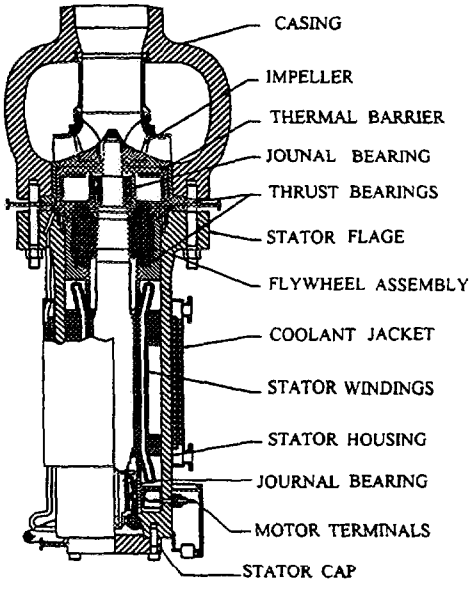


Figure 4.4-4 EP 1000 reactor coolant pump - Reliable canned motor design

The motor developed for the AP-600 is a 60Hz design, but preliminary design studies have estimated the overall parameters and envelope size for the 50Hz design required for the EP 1000. The canned motor pump is a highly reliable unit used extensively in nuclear and fossil applications. It has no seals and does not require a seal water system, i.e., continuous charging pump operation is not required and the chemical and volume control system is simplified. Besides, with the absence of seals, seal failure LOCAs are eliminated, and this is a significant safety enhancement, since seal failure LOCAs are a major industry issue. Maintenance is also enhanced, since seal replacement is not necessary.

One modification of the EP 1000 pumps from commercial and marine canned motor pump practice is the use of a flywheel to increase the pump rotating inertia. The increased inertia provides a slower rate-of-flow coastdown to improve core thermal margins following loss of electric power.

4.4.2.5 Reactor auxiliary systems

Residual heat removal system

The normal residual heat removal system (RNS) consists of two separate mechanical trains of equipment. Each train consists of one pump and one heat exchanger. In addition, RNS comprises piping, valves and intrumentation necessary for correct system operation. The RNS is located completely inside the containment.

The RNS is designed to perform the following major functions:

- Remove heat from the core and RCS during normal coldowns and during refuelling operations;
- Provide connections to the RCS that facilitate draining the RCS down to mid-loop level during refuelling operations;
- Provides RCS and refuelling cavity purification flow to the Chemical and volume control system during refuelling operation;
- Post accident recovery by providing heat removal from the core and RCS following successfull mitigation of an accident by the passive core cooling system;
- Provides low pressure makeup to the RCS following recovery from inadvertent actuations of Automatic depressurization system; and
- Provides low temperature overpressure protection for the RCS during refuelling, startup and shutdown conditions

The RNS is designed to perform its functions in a very reliable and failure tolerant manner. The reliability is achieved with the use of highly reliable and redundant equipment and with a simplified design but not by the use of safety class equipment or technical specifications. The RNS is not a safety system; however, the system is designed in accordance with equipment class D standards because it acts to prevent actuation of the passive safety systems and because it is taken credit for in the PSA studies

Chemical and volume control system

The chemical and volume control system (CVS) consists of regenerative and letdown heat exchangers, demineralizers and filters, makeup pumps, tanks, and associated valves, piping, and instrumentation. The principal functions of the CVS includes:

- Maintain RCS fluid purity and activity level within acceptable limits;
- RCS inventory control and makeup;
- Chemical shim and chemical control; and
- Provides borated makeup to auxiliary equipment.

The CVS provides some functions that are needed for the continued normal operation of the plant, and it is designed to satisfy these functions in a very reliable and failure tolerant manner. This reliability is achieved through the use of redundant equipment (pumps, filters and demineralizers) and with a simplified design.

The CVS is not a safety system, and is therefore not required to mitigate design basis events. For these events, the makeup and boration function is satisfied by the passive core cooling system (PXS). However, the system is designed in accordance with equipment class D standards because its operation prevents actuation of the passive safety features and because it is taken credit for in probabilistic safety assessments (PSAs).

The safety functions provided by the CVS include containment isolation of the lines penetrating the containment, termination of RCS boron dilution, and preservation of RCS pressure boundary, including isolation of normal CVS letdown from the RCS.

Startup feedwater system

The startup feedwater system (STS) consists of two motor driven centrifugal pumps with automatic connection to diesel power, two miniflow control valves, a suction control valve and associated piping to provide feedwater flow to the steam generators.

The function of the STS is to provide a source of feedwater to the steam generators under low flow conditions such as startup, hot standby, shutdown conditions or when the main feedwater pumps may be unavailable or oversized for the required flow rate. The startup feedwater pumps are designed to deliver feedwater to the steam generators using the deaerator as the normal suction source. The pumps start automatically following anticipated system transients as well as events in which the main feedwater system is unavailable. A single startup feedwater pump is sized to provide sufficient water to remove core or decay heat via the steam generators and prevent PRHR (Passive RHR) actuation.

Cooling water systems

Service water system (SWS). The SWS is a non safety related system that has no safety related design bases since it is not required to perform any safety related function to support accident mitigation or to establish and maintain a safe shutdown condition following DBA. The SWS provides the defense in depth function of removing heat from the component cooling water system for all modes of operation and from the NRS during plant shutdown.

The SWS transfers heat to the ultimate heat sink from the following components during all modes of plant operation and power generation:

- CCS component cooling heat exchangers;
- Turbine building closed cooling system (TCS) heat exchangers; and
- Central chilled water system (VWS) chillers.

The system consists of two 100% capacity sea water trains that can be cross-connected in a variety of ways in the event of component failures. Each train includes one sea water pump, one primary flow strainer, a dedicated supply header, one set of associated heat exchangers, one discharge header, one motor cooling flow path with cyclone separator, and associated train piping, valves control and instrumentation.

Component cooling water system (CCWS). The component cooling water system is a non-safety, closed loop cooling system designed to transfer heat from various sources to the SWS during all plant operating modes. More specifically, the CCWS provides the following cooling functions:

- Removes heat from various components needed for normal plant operation (included RCPs);
- Removes core decay heat and sensible heat for normal reactor shutdown and cooldown.

The system is arranged into two mechanical trains. Each train contains one pump and one heat exchanger. The two trains take suction from the same return header. Each pump discharge into the respective heat exchanger but a cross-connection is available to allow each of the two pumps to feed both heat exchangers. The discharge of each heat exchanger is routed to a common header.

Spent fuel pit cooling system. The spent fuel pit cooling system (SFS) is designed to perform the following major functions:

- Remove heat from the water in the spent fuel pit during all modes of operation to maintain a desired water temperature;
- Provide purification and clarification of the spent fuel pit water during all modes of operation;
- Provide purification of the refuelling cavity during refuelling operation; and
- Provide purification of the In-containment-water-storage-tank during all modes of operation.

The SFS consists of two mechanical trains of equipment. Each train is designed to handle 330 m^3/h, and it can be operated independently from the other train. Each train consists of one pump, one heat exchanger, one demineralizer and one filter. In addition, the SFS comprises piping, valves and instrumentation necessary for correct system operation. The two trains share common suction and discharge headers.

In accordance with the EUR requirements, the SFS is sized for the maximum heat load due to the decay heat generated by the accumulation of 15 years of MOX spent fuel (that constitutes the 50% of the core assemblies) plus 10 years of UO_2 spent fuel plus one full core offload (96 MOX assemblies plus 97 UO_2 assemblies) placed in the pool beginning at 100 hours after shutdown and completed 48 hours later.

HVAC systems - passive annulus filtration system. The Passive annulus filtration systems (PAFS) is part of the EP 1000 HVAC systems. It is designed to function following a severe accident to limit the offsite dose. The EP 1000 is equipped with a secondary containment. For the steel containment reference configuration, the secondary containment is defined as the structure which confines the penetration area to collect leakages through the penetrations that constitute the major source of containment leakage.

The PAFS is designed to perform the following major functions:
- Contribute to the limitation of the offsite dose to the value defined by site parameters; and
- Maintain a negative pressure in the annulus penetration (secondary containment).

Initial evaluations indicate the EP 1000 radioactive releases to the environment will be low, in compliance with the EUR safety targets for Design basis accidents (DBAs) without operation of the PAFS. Therefore, the PAFS is provided to fulfil a level F2 safety function which in the EUR Rev. B is defined as, "to ensure that the releases are kept within the targets set for DEC (design extension conditions)".

The PAFS is connected to the middle and lower annulus of the secondary containment. It consists of two mechanical trains of equipment. Each train consists of one HEPA filter, one eductor and a compressed air storage. The motive force of the eductor is the compressed air stored in tanks, having a capacity per train to support the function of the PAFS for the first 24 hours after a DEC accident. The capacity of both trains, used one after the other, should be able to perform the PAFS function for a period of 72 h.

Although the PAFS does not need to function following a DBA, an Annulus filtration system is recommended to meet the acceptable releases level for economical impact target. Since the system is not required to function during a DBA, the use of an active system versus a passive system will be investigated later in the Phase 2 activities.

Other major auxiliary systems

Other major auxiliary systems are the:
- Chilled water system;
- Hydrogen control system;
- Liquid radwaste system;
- Gaseous radwaste system;
- Steam generator blowdown system;
- HVAC systems;
- Fire protection system;
- Primary sampling system.

4.4.2.6 Operating characteristics

The plant control scheme is based on the "Reactor follows plant loads". A grid fluctuation can be compensated for through turbine control valves in case of frequency drop. A decrease in pressure at the turbine would require an increase in reactor power.

While the plant performance criteria reported in the Revision B of the EUR have not yet been evaluated in detail, a Phase 1 evaluation of the EUR Rev. A did not indicate any major issue other than the grid frequency change.

It is already known that some of the revision B grid requirements such as the starting and loading times (i.e., 24 hours from cold shutdown (less than 60°C) to hot shutdown at full pressure, and 20 hours from hot zero power to cold shutdown), are met by the EP 1000 plant.

The EPP reference plant design is based on the AP-600 and SPWR plants which are 60 Hz designs. The implementation of 50Hz equipment is not expected to cause any large consequences, however, and the EP 1000 is expected to withstand, among others, the following operational occurrences, defined in the EUR requirements, without resulting in a reactor trip or actuation of the safety-related passive engineered safety systems:

- +/- 5% minute ramp load change within 15% and 100% power;
- +/- 10% step load change within 20% and 100% power;
- 100% generator load rejection;
- Frequency range for synchronization: 48 - 52 Hz; and
- 100-50-100 power level daily load follow.

The logic and the setpoints for the EP 1000 nuclear steam supply system (NSSS) control systems will be developed to meet the above operational transients without reaching any of the protection system setpoints.

Moreover, the EP 1000 is designed to meet additional requirements for core operation such as the capability to use up to 50% MOX fuel.

4.4.3 Description of turbine generator plant system

Activities on the turbine generator system will be developed later in the EPP programme.

4.4.4 Instrumentation and control systems

4.4.4.1 I&C Design concept, including control room

I&C structure

The instrumentation and control systems provide protection against unsafe reactor operation during steady state and transient power operations. They initiate selected protective functions to mitigate the consequences of design basis events (DBEs).

The EPP I&C will be designed in a similar way as the I&C of the AP-600. Advanced, micro-processor based instrumentation and control (I&C) systems contribute to the overall plant safety by simplifying and enhancing plant operation and maintenance. A digital, multiplexed control system takes the place of hard-wired analog controls and cable spreading rooms, accounting for a significant reduction in cable usage. The I&C components feature built in diagnostics and board level repair, making restoration of operability at most faults in the system a matter of replacement of a printed circuit cards or instrument module.

The EPP instrumentation and control architecture includes the following major systems:

- The Protection and safety monitoring system (Section 4.4.4.2).

- The plant control systems which perform non safety related instrumentation and control functions using both discrete (on/off) and modulating (analog) type actuation devices.
  The plant control systems contains instrumentation and equipment to change reactor power, control pressurizer pressure and level, control feedwater flow and perform other plant functions associated with power generation.
- The diverse actuation system is a non-safety-related, diverse system that provides an alternate means of initiating reactor trip and selected engineered safety features and provides plant information to the operator.
- The data display and acquisition system provides the equipment used for processing data that will result in non-safety-related alarms and displays for both normal and emergency plant operation, generating these displays and alarms, providing analysis of plant data, providing plant data logging and historical storage and retrieval, and providing operational support for plant personnel.
- The in-core instrumentation system provides a three dimensional flux map of the reactor core. This map is used to calibrate neutron detectors used by the protection and safety monitoring system, as well as to optimize core performance. A secondary function of the in-core instrumentation system is to provide the protection and safety monitoring system with the thermocouple signals necessary for the post-accident inadequate core cooling monitor. The in-core instrument assemblies house both fixed in-core detectors and core exit thermocouples.
- The special monitoring system that consists of specialized subsystems that interface with the instrumentation and control architecture to provide diagnostics and long-term monitoring functions.
- The operation and control center system that includes the main control room and the distributed computer system.

Main control room

The EPP main control room will be designed in a similar way as that of the AP-600, which is a state-of-the-art design meeting and defining the current standards for licensability and operator interface.

The EPP control room consists of two independent sets of controls (operator and supervisor), plus an integrated wall panel information system and a set of dedicated controls and displays. Datalinks are used to communicate between the control stations and the protection system. The alarm system which categorizes and displays alarms, is interfaced to a modern display system that makes it easy for operators to analyze the status of the whole system while sitting at their work positions. The operator can select any function and the relevant display will appear on the screen. The operator can interrogate that function for any information.

4.4.4.2 Reactor protection and other safety systems

The protection and monitoring system (PMS) includes the reactor trip subsystem and the engineered safety feature actuation subsystem. The related cabinets, sensors, and the reactor trip switchgear are four way redundant. This redundancy permits the use of automatically invoked bypass logic so that a division or individual channel out of service can be accomodated by the operating portion of the protection system reverting to a two-out-of-three logic from a two-out-of-four logic. The protection and safety monitoring system provides the safety-related functions necessary to control the plant during normal operation, to shut down the plant, and to maintain the plant in a safe shutdown condition. The PMS controls safety-related components in the plant that are operated from the main control room or remote shutdown work position.

In addition, the PMS provides the equipment necessary to monitor the plant safety-related functions during and following an accident as required by US NRC Regulatory Guide 1.97.

4.4.5 Electrical systems

The two main electrical systems are the main AC power system and the DC power system. The main AC power system is a non class 1E system and non safety related. The DC power system consists of

two independent systems: the class 1E DC system and the non class 1E DC system. Each DC system consists of ungrounded station batteries, DC distribution equipment, and uninterruptible power supply (UPS).

4.4.5.1 Operational power supply systems

The main AC power system (ECS) consists of two power systems, the off-site power system and the on-site power system. The off-site power system serves to transmit the generated power to the grid and to the plant auxiliary and service loads. During plant startup, shutdown, or outage the offsite power system supplies power to the plant auxiliaries and service loads. If the off-site power sources are unavailable, on-site power sources supply loads which have to remain permanently operational including when the unit is shut down (permanent auxiliaries).

The on-site distribution system is arranged in two hierarchy levels:

- Distribution system for power supply of auxiliary loads (unit operation loads) and permanent auxiliary (unit shutdown loads); and
- Distribution system for power supply of safety loads (IDS).

Offsite AC power system

The off-site AC power is not required for plant safety. The normal power sources for the plant auxiliary AC loads are provided from the 22 kV isophase generator buses through the unit auxiliary transformers. When the main generator is not available, the plant auxiliary power comes from the switchyard by backfeeding through the main transformer and the auxiliary transformers. This is the preferred power supply.

In the event that both alternative sources are not available, the auxiliary loads will be powered through the startup transformer which is connected to another off-site transmission. Each unit auxiliary transformer supplies about 50% of the auxiliary loads. The start-up transformer has the capacity to supply 100% of the loads. The unit auxiliary and the start-up transformers are all of three winding type.

The protective relaying and metering required for the transformer and the other equipment in the transformer area is located in the turbine building. The necessary power supply sources of 400/230 V AC and 110 V DC to the equipment are provided in the turbine building.

On-site standby power system (ZOS)

The on-site standby power system of the EP 1000 is designed to supply AC power to the selected plant non-safety loads in the event of main generator trip concurrent with the loss of off-site power.

The power is provided to selected loads that support defence-in-depth functions and protect the owner investment. Also, the ZOS is designed to be available to supply plant power during maintenance of the off-site power supply system when the plant is in cold shutdown conditions. Operation of the ZOS is not required to ensure nuclear safety. The on-site standby power source system is of non-safety class, and as such, it is not classified as a Class 1E system.

The ZOS consists of two standby diesel generators, one per reactor related bus ES1 and ES2, to provide power supply to the plant permanent non-safety loads in the event of a turbine trip with loss of off-site power. Each diesel generator unit is an independent self-contained system, complete with all necessary support sub-systems.

Following a loss of off site power, each diesel generator shall be capable of automatically starting, accelerating to rated speed, attain rated voltage and be ready to supply the design loads within twenty (20) seconds from the receipt of a start signal. Automatic load sequencing is used to facilitate orderly loading of the selected permanent non-safety loads to aid operator response to the loss of normal power event.

Non-Class 1E DC and UPS system

The non-Class 1E DC and UPS systems consists of electrical power supply and distribution equipment that provides DC and uninterruptable AC power to the plant non-Class 1E DC and AC loads that are critical for safe plant operation and investment protection.

The non-Class 1E DC and UPS system consists of two subsystems representing two separate power supply trains and one separate 250 VDC subsystem for motor drives.

The non-Class 1E DC distribution subsystem bus has provision to allow the connection of a permanent installed spare non-Class 1E battery charger, if the ordinary battery charger is unavailable due to maintenance, testing or repair.

4.4.5.2 Safety-related systems

Class 1E DC and UPS system (IDS)

The Class 1E DC and UPS system (IDS) provides uninterruptable power for the plant instrumentation, control, monitoring and other vital functions that are required for plant startup, normal operation and normal/emergency shutdown of the plant. The IDS is a Class 1E system. It is required to perform safety-related functions for the plant safe shutdown. In the event of a total loss of off-site and on-site AC power sources, DC batteries constitute the sources of electrical power for operation of the required DC and AC instrument uninterruptable power supply (UPS) loads. In addition, the IDS system provides power to the normal and emergency lighting in the main control room (MCR) and in the remote shutdown area (RSA).

The IDS is designed to ensure that sufficient power is available for the safety-related equipment required for safe shutdown of the plant, and for mitigation and control of accident conditions in the plant.

The safety design basis for the system includes the following. The IDS is designed such that the critical plant loads required for safe plant shutdown and monitoring are powered by batteries, when all the other on-site and off-site AC power sources at the plant are lost and cannot be recoverd for a period of 72 hours. System operability is based on ensuring that the plant can be maintained in the shutdown condition for an extended period of time and that sufficient instrumentation is available for monitoring and maintaining the status of the plant. After 24 hours, no control functions are supported. Monitoring and lighting are supported for 72 hours.

The IDS allows for Class 1E 110 V DC and Class 1E UPS 400/230 V AC power distribution.

The Class 1E 110 V DC subsystem is divided in four independent divisions, each one being supplied by battery chargers and dedicated batteries. Each division has two battery systems. One system is designed for 24 hour battery bank and the other system as 72 hour battery bank.

All four independent divisions have individual Class 1E UPS subsystems providing power at 400/230 V AC. There are two UPS in each division, one for 24 hour loads and the other for 72 hour loads. The UPS are supplied with power from the corresponding battery banks.

4.4.6 Safety concepts

4.4.6.1 Safety requirements and design philosophy

The basic EP 1000 safety philosophy is based on utilization of inherent margins (e.g. larger volumes and water inventory, lower power density, negative power and temperature reactivity coefficients) to limit system challanges.

Consistent with current practice,active systems are used as first level of defence against more probable events. These non-safety systems include, among others, the chemical and volume control system

(CVCS), Startup feedwater system, the Normal residual heat removal system, and the Spent fuel pit cooling system (SFS). While these systems are not required to operate during acident conditions, they are designed to be highly reliable. For this reason, non-safety AC power sources are available to power these systems.

As a second line of defence the EP 1000 design includes passive safety features described in the following.

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

Passive safety systems

The EP 1000 uses passive safety systems to further enhance plant safety and to satisfy US NRC safety criteria and EUR requirements. The use of passive safety systems has provided significant and measurable improvements in plant simplification, safety, reliability, and investment protection. The passive safety systems require no operator actions to mitigate design basis accidents. These systems use only natural forces such as gravity, natural circulation, and compressed gas to make the systems work; no pumps, fans, diesels, chillers, or other active equipment are used. A few simple valves align and automatically actuate the passive safety systems. To provide high reliability, these valves are designed to actuate to their safe positions upon loss of power or upon receipt of a safeguards actuation signal. However, they are also supported by multiple, reliable power sources to avoid unnecessary actuations.

The EP 1000 passive safety-related systems include:

- The passive core cooling system (PXS);
- The passive containment cooling system (PCS);
- The main control room habitability system (VES); and
- Containment isolation.

These passive safety systems provide a major enhancement in plant safety and investment protection as compared with conventional plants. They establish and maintain core cooling and containment integrity indefinitely, with no operator or AC power support requirements. The passive systems are designed to meet the US NRC single-failure criterion, and probabilistic safety assessments (PSAs) are used to verify their reliability. These passive safety systems are also designed to satisfy other US NRC requirements and rules, including Three Mile Island lessons learned, Standard Review Plan, Regulatory Guides, and unresolved and generic safety issues. A description of each of the passive safety systems follows below, with an overview of the plant simplification they provide.

Passive core cooling system. The PXS protects the plant against reactor coolant system (RCS) leaks and ruptures of various sizes and locations. The PXS provides the safety functions of core residual heat removal, safety injection, and depressurization. Safety analyses (using US NRC-approved codes) demonstrate the effectiveness of the PXS in protecting the core following all sizes of RCS break events. Even for breaks as severe as the 8-inch vessel injection lines the core remains covered. The PXS provides approximately a 400°F (220 °C) margin to the maximum peak clad temperature limit for the double-ended rupture of a main reactor coolant pipe.

Safety injection and depressurization. The PXS uses four passive sources of water to maintain core cooling through safety injection. These injection sources include the core makeup tanks (CMTs), the core reflood tanks (CRTs), the accumulators, and the in-containment refuelling water storage tanks (IRWST). These injection sources are directly connected to two nozzles on the reactor vessel so that no injection flow can be spilled for the larger break cases.

The CMTs provide makeup at any RCS pressure, using gravity to provide injection flow. These tanks are located inside the containment, above the RCS loops. The CMT pressure is equalized with the RCS through a line connecting the top of the CMTs to the RCS. The elevation head is sufficient to overcome the small pressure loss in the injection line. The CMTs are normally full of borated water and isolated by two parallel "fail-open" air-operated globe valves. The isolation valves open when the RCS

pressure or level drops to abnormal levels. The tanks are sized to provide extended makeup to the RCS in the case of minor leakage.

For larger leaks, additional water is provided by the accumulators and core reflood tanks (CRTs) which inject water pressurized by compressed nitrogen. The two accumulators tanks operate passively when the RCS pressure drops below the normal gas pressure of 700 psig. The two CRTs operate passively when the RCS pressure drops below the normal gas pressure of 200 psig. The gas pressure forces open check valves that normally isolate the accumulators and CRTs from the RCS. The accumulators and CRTs are sized to respond to the complete severance of the largest RCS pipe, by rapidly refilling the vessel downcomer and lower plenum.

Long-term injection water is provided by gravity from the IRWST, which is located in the containment just above the RCS loops. Normally, the IRWST is isolated from the RCS by check valves. This tank is designed for atmospheric pressure, and the RCS must be depressurized before injection can occur. The depressurization of the RCS is automatically controlled to reduce its pressure to about 12 psig; at which point, the head of water in the IRWST overcomes the low RCS pressure and the pressure loss in the injection lines. The PXS provides for depressurization using its automatic depressurization system (ADS), composed of four stages to permit a relatively slow, controlled RCS pressure reduction. The first three stages are connected to the pressurizer and discharge through spargers into the IRWST. The three fourth-stage depressurization paths are connected to the hot legs, and discharge through redundant isolation valves to the containment. The initial ADS stages are actuated by the CMT water level, using 2-out-of-4 logic to ensure reliability and to prevent spurious actuations. All of the valves use existing nuclear-grade valve body and operator designs, extensively tested to ensure operability in this application.

During a LOCA, the initial volume of water in the IRWST provides injection for at least 6 hours. As it empties, the containment water level rises above the RCS loop level. This level is sufficient to force water to drain through a screen and check valves back into the RCS where it is turned into steam by core decay heat. The steam is vented to the containment through the ADS valves and the break where it condenses on the inside of the steel containment vessel. The condensate drains down into the IRWST and again becomes available for injection into the RCS.

Passive residual heat removal. The PXS includes two identical passive residual heat removal heat exchangers (PRHR Hxs) which are connected through inlet and outlet lines to two of the RCS loops. The PRHR HXs protect the plant against transients that upset the normal steam generator feedwater and steam systems, and satisfy the US NRC safety criteria for loss of feedwater, and feedwater and steam line breaks using single failure assumptions approved by US NRC safety analysis codes.

The IRWST provides the heat sink for the PRHR HXs. The IRWST water volume is sufficient to absorb decay heat for more than 1 hour before the water begins to boil. Once boiling starts, steam passes to the containment where it condenses on the inside of the steel containment vessel and, after collection, drains by gravity back into the IRWST. The PRHR HXs, along with the passive containment cooling, provides indefinite decay heat removal capability with no operator action required. The operator is provided with the capability of controlling the PRHR HX rate such that the RCS temperature can be controlled, if desired. This capability also allows for control of the rate of RCS cooldown.

Passive containment cooling system. The passive containment cooling system (PCS) provides the safety-related ultimate heat sink for the plant. As demonstrated by computer analyses and extensive test programs, the PCS effectively cools the containment following an accident such that the design pressure is not exceeded and the pressure is rapidly reduced. For single steel containments, the steel containment vessel itself provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Steel containment vessels of similar size have been used on operating PWRs. Heat is removed from the containment vessel by continuous natural circulation flow of air. During an accident, the air cooling is supplemented by evaporation of water. The water drains by gravity from a tank located on top of the containment shield building. Two normally closed fail-open butterfly valves are opened to initiate the

water flow. The water tank is sized for 72 hours of operation, after which time the tank is expected to be refilled so that the low containment pressure achieved after the accident (1/2 design pressure in 24 hours) can be maintained. If the water is not resupplied, the containment pressure will increase, but the peak is calculated to be less than design pressure even after two weeks without operator support actions.

The EP 1000 will have a significantly reduced frequency of release of large amounts of radioactivity following a severe accident core damage scenario. Analysis shows that with only the normal PCS air cooling, the containment pressure will stay well below the predicted failure pressure. Other factors include improved containment isolation and reduced potential for LOCAs outside the containment. This improved containment performance supports the technical basis for simplification of the off-site emergency planning.

Main control room habitability system. The EP 1000 main control room habitability system (VES) provides fresh air, cooling, and pressurization to the main control room (MCR) following a plant accident. Operation of the VES is automatically initiated upon receipt of a high MCR radiation signal, which isolates the normal control room ventilation path and initiates pressurization. Following system actuation, all functions are completely passive.

Containment isolation. EP 1000 containment isolation is significantly improved over that of current-day PWRs. One major improvement is the large reduction in the number of penetrations. Furthermore, the number of normally open penetrations is reduced by 60 percent. For example, the chemical and volume control system (CVS) letdown penetration is normally closed because the CVS purification occurs in a high-pressure loop, inside the containment. Also, there are no penetrations required to support post-accident mitigation functions (the canned motor reactor coolant pumps do not require seal injection, and the residual heat removal and safety injection features are located entirely inside containment).

Safety systems simplification. The passive safety systems are significantly simpler than typical PWR safety systems since they contain significantly fewer components, reducing the required tests, inspections, and maintenance, require no active support systems, and their readiness is easily monitored.

A major EP 1000 plant simplification is the elimination of the traditional safety-related containment spray system. This system is normally required to remove airborne particulates and elemental iodine releases to the containment atmosphere following a core degradation accident in a current-day plant. For the EP 1000, removal of airborne activity is achieved by natural processes such as sedimentation and deposition that do not depend on sprays.

4.4.6.3 Severe accidents (Beyond Design Basis accidents)

The assessment of the EP 1000 performance against severe accidents will be performed in accordance with the European Utilities Requirements; the general approach involves identifying the sequences to be reduced in probability below the credibility threshold and those to be mitigated.

According to the EUR, the assessment of the Design extension conditions (DEC) in addition to the Design basis accident (DBA) is the preferred method for giving consideration to the complex sequences and severe accidents at the design stage without including them in the Design basis conditions. The assessment of the DEC permits the definition and evaluation of the Design extension measures (DEM) to prevent core melting or mitigate the consequences of accident sequences such as:

- Complex sequences which involve failures beyond those considered in the deterministic design basis
- Severe accidents, both to prevent early and delayed containment failure and to minimize releases for the remaining conditions that go beyond the design basis conditions (DBC).

The Design extension concept makes use of probabilistic methods as one way to identify the need for the implementation of measures, including upgraded or additional equipment or accident procedures for complex sequences and severe accidents that provide a significant contribution to the core damage frequency and/or frequency of exceeding acceptable limiting releases.

A preliminary probabilistic safety assessment (PSA) shows that the EP 1000 plant has a level of risk similar to AP-600 and meets and exceeds the design goals specified in the EUR. Preliminary evaluations provide a core damage frequency of 2.3×10^{-7} per year for internal events at power conditions. In particular, the analyses have shown that many of the events that, in the past, were leading contributors to the risk of nuclear power plants, such as interfacing LOCAs, are not equally significant for the EP 1000.

Severe accident analyses involving core melt have been performed with the MAAP 4.0 code to evaluate the potential for significant releases to the environment. The results of the analysis have pointed out that the EP 1000 passive containment system provides a significant improvement in terms of safety and reliability compared to conventional containment systems.

4.4.7 Plant layout

4.4.7.1 Buildings and structures, including plot plan

The site plan for a single-unit EP 1000 standard plant is shown in Figure 4.4-5. The power block complex consists of six principal building structures: the nuclear island, the turbine building, the annex building, the diesel generator building, and the radwaste building. Each of these building structures is constructed on individual basemats. The nuclear island consists of the containment building, and the auxiliary building, all of which are constructed on a common basemat.

The EP 1000 plant design helps to minimize the construction schedule and cost, and meets the safety, operational, maintenance, and structural criteria. The most direct means for helping to minimize the construction schedule and the capital cost is to minimize the building volume and the bulk quantities while satisfying all layout criteria, specifically the criteria related to maintenance activities and maintenance space allocation. The EP 1000 arrangement criteria provide a significant increase in maintenance space allocation in many areas as compared with most current-day plants.

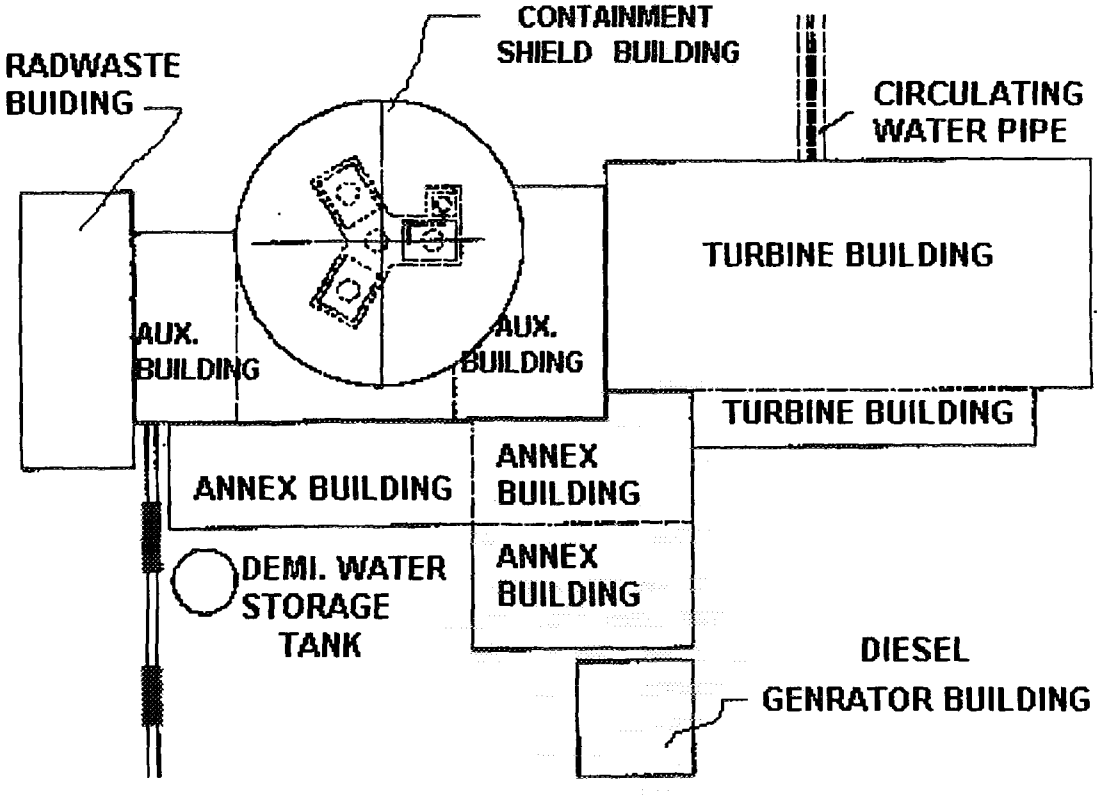


Figure 4.4-5 Site plan for a single-unit EP 1000 standard plant

The following key design features are incorporated into the plant configuration and contribute to a plant arrangement that meets the cost and schedule objectives:

- Utilization of the integrated protection system, the advanced control room, the distributed logic cabinets, multiplexing, and fiber optics, significantly reduces the quantity of cables, cable trays, and conduits. These bulk quantity reductions reduce the building volume requirements, and that in turn minimizes the construction cost and schedule.
- Application of the passive safeguards systems replaces and/or eliminates many of the conventional mechanical safeguards systems that are typically located in the Seismic Category I buildings in the current generation of PWR plants. The elimination of these mechanical safeguards systems and their associated pumps, valves, and piping results in a major reduction in the required Seismic Category I building volume and a major reduction in the capital cost and construction schedule.
- The passive safeguards systems eliminate the requirement for two large Class 1E, Seismic Category I, emergency standby diesel generators, and the Class 1E safeguard busses that are required in the current generation of PWR plants. The elimination of the Class 1E diesel generator results in a major reduction in the associated Class 1E cables, cable trays, and large circuit breakers.
- A key feature of the EP 1000 plant configuration is the stacked arrangement of the Class 1E battery rooms, the dc switchgear rooms, the integrated protection system rooms, and the main control room. This stacked arrangement eliminates the need for the upper and lower cable spreading rooms that are required in the current generation of PWR plants. This elimination significantly reduces the required Seismic Category I building volume as well as the difficult construction task associated with cable spreading rooms.

Another key EP 1000 plant arrangement feature is the location of the IRWST inside the containment building. This IRWST location eliminates the need for a dedicated, missile-protected, Seismic Category I building.

The EP 1000 plant arrangement is designed so that operational and maintenance activities can be accomplished in an efficient manner. A key objective is to provide confidence that the utility needs and expectations for operability and maintainability are appropriately included in the design. The plant arrangement contains a mixture of conventional and unique features that facilitate and simplify operational and maintenance activities.

The EP 1000 plant arrangement provides separation between safety and non-safety equipment, radioactive and non-radioactive equipment, and mechanical and electrical equipment. These separation features in the equipment layout also facilitate maintenance. The radioactive equipment and piping are arranged and shielded to minimize radiation exposure, which is beneficial to the operation and maintenance activities.

EP 1000 general arrangement design basis

The nuclear island layout is based on a replication of the containment building layout of the Westinghouse SPWR with some minor changes and a general auxiliary buildings arrangement, following the general concepts of the AP-600. Hence, the reference design of the SPWR/AP-600 (AP-600's description in Chapter 5) is modified only to the extent required to accomodate the following deviations:

- the changes in system design and size that are incorporated in the EP 1000;
- the EUR protection against aircraft crash which requires that a standard design either provide protection against aircraft crash or can cope with the incorporation of such a protection without jeopardizing the layout and the functional specification of the plant;
- European seismic requirements for structures and components;
- External pressure waves requirements for containment structures;
- The incorporation of a third compartment for main steam and feedwater lines in the AP-600 auxiliary building configuration which had been initially developed for a two-loop reactor coolant system;
- The two containment designs considered for the EP 1000 - single steel containment (SSC) and double concrete containment (DCC) - while ensuring a common layout for the reactor building, minimizing

the impact on auxiliary building plant layout and the provisions that are needed to achieve this compatibility so as not to jeopardize cost-effectiveness of either option;

- the interface between the SPWR containment building which was designed for a "wrap-around" configuration of the auxiliary building and an AP-600 style auxiliary building which provides for a non-symmetric configuration.

Seismic behaviour

All structures and equipment required to fulfill Level Fl Safety Functions (as defined in EUR Rev. B) during or after an earthquake shall be Seismic Category 1, and qualified to withstand the effect of a safe shutdown earthquake (SSE = Design Basis Earthquake (DBE) in the EUR), i.e., to remain structurally intact, leaktight in case of fluid retaining equipment, and functionally operable to the extent required by its safety role.

Structures and equipment required to fulfill Level F2 Safety Functions during or after an earthquake shall be identified, on a case by case basis, to establish the need for seismic qualification or other means of ensuring its capability to withstand earthquake-induced effects to the extent required by its contribution to nuclear safety. These means may include comparison with comparable structures or equipment which experienced actual earthquake conditions. Such equipment shall be Seismic Category II.

In addition, Non-Seismic-Category-I components and structures, of which the failure in SSE conditions could impair the correct functioning of Seismic Category I equipment, shall be assigned to Seismic Category S. (EUR Rev. B, Chapter 2.4 - Section 2.4.4.2.1.)

Seismic input

The effects of earthquakes shall, in accordance with EUR rev. B, be taken into account by designing the plant against a suite of vibratory ground motions which together define the Safe Shutdown Earthquake (SSE) (or design basis earthquake (DBE) in the EUR). It shall be demonstrated that, when subjected to the defined loading, the reactor can be brought into and maintained in a safe shutdown state.

The horizontal free field Peak Ground Acceleration of the DBE (= SSE) of the EUR has been established as 0.25 g. This applies to the standard design. For licensing purposes, the Owner will determine, on the basis of the seismicity and geology of the specific site under consideration, the parameters of the site-specific SSE. It should be noted that the standard design is not intended to envelope all possible combinations of national regulations and site conditions. The Owner may specify one or more additional vibratory ground motions corresponding to lower levels of earthquake than the SSE in order to satisfy investment protection or local licensing requirements.

4.4.7.2 Reactor building(Section 4.4.7.3.)

4.4.7.3 Containment

The containment building consists of the containment and all structures within the containment. It is an integral part of the overall containment system with the functions of containing the release of airborne radioactivity following postulated design basis accidents and providing shielding for the reactor core and the reactor coolant system during normal operations. Representative general arrangement drawings of the containment building are shown in Figures 4.4-6 and 4.4-7 on next page.

Two containment designs are considered for the EP 1000 - single steel containment (SSC) and double concrete containment (DCC). In the following, only the SSC reference configuration will be discussed.

The containment vessel is an integral part of the passive containment cooling systems (PCS). The containment vessel and the PCS are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents. A Seismic Category I structure, the containment vessel is a free standing steel cylinder, 46 meters in diameter and 67.8.

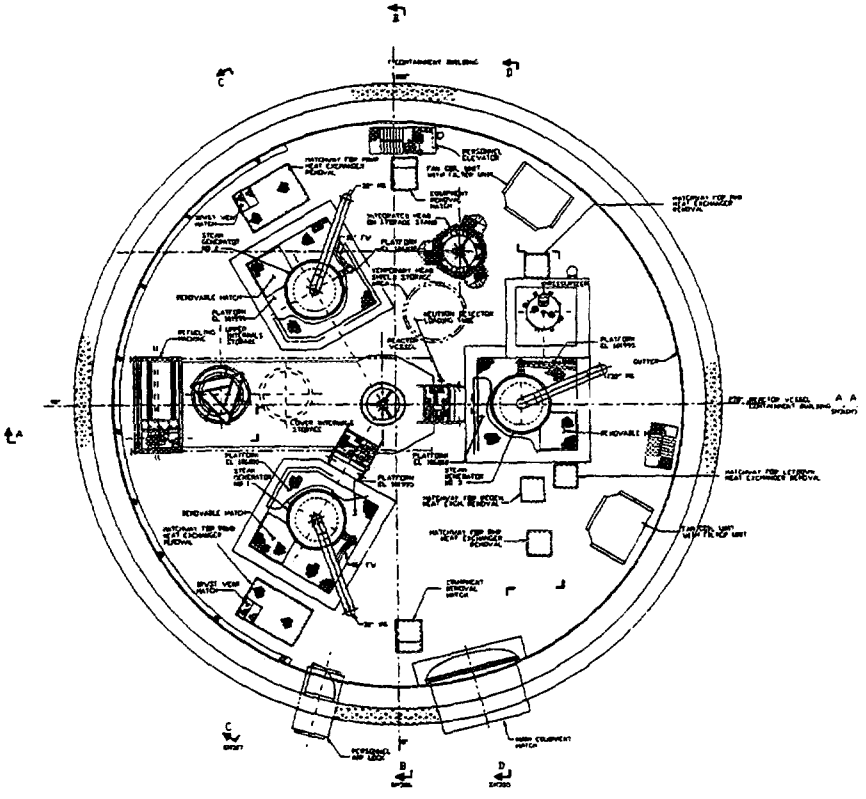


Figure 4.4-6 General containment arrangement; plan view of operating floor

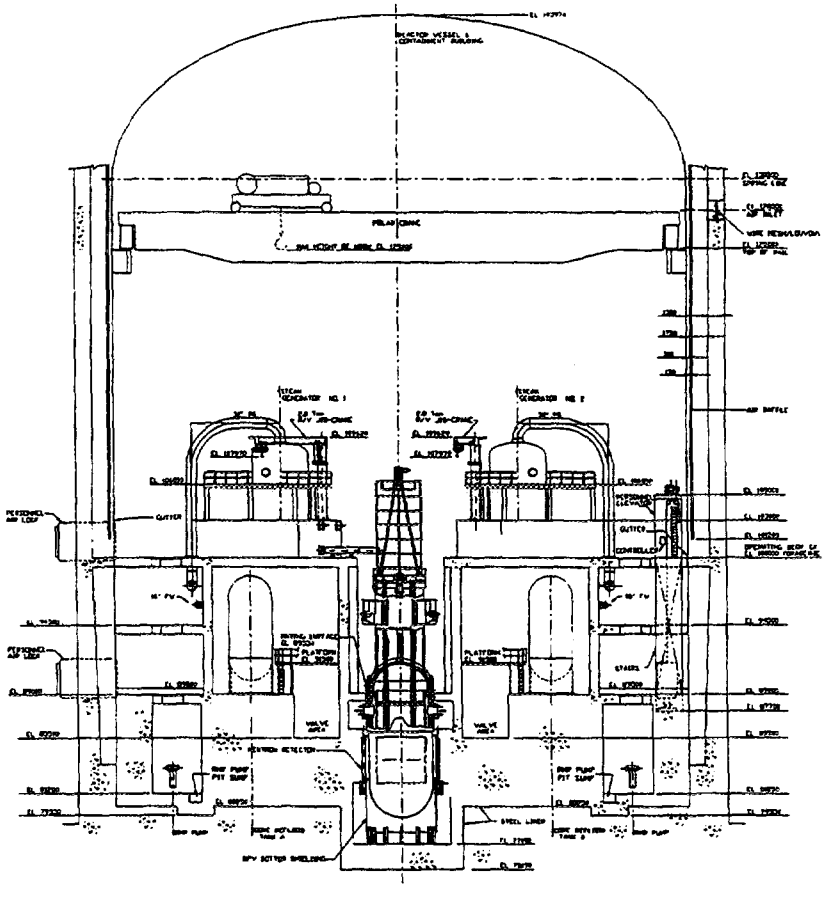


Figure 4.4-7 General containment arrangement; elevation view at Section B-B

meters from the containment sump to the inside containment top head. It is surrounded by a Seismic Category I reinforced concrete shield building

There are three floor elevations (grade access maintenance floor and operating deck) and ten equipment compartments within the containment building. Floor gratings are provided for access to equipment at other elevations. The principal systems located within the containment building are the reactor coolant system (RCS), the passive core cooling system (PXS), the normal heat removal system (RNS), and the chemical and volume control system (CVS).

4.4.7.4 Turbine building

Although not specifically developed during the Phase 1 of the EPP programme, this section provides an overview of the SPWR/AP-600 turbine building that will be the basis for the development of the EP 1000 turbine building.

The turbine building houses the main turbine, generator, and associated fluid and electrical systems. It provides weather protection for the laydown and maintenance of major turbine/generator components.

The turbine building also houses the makeup water purification system. No safety-related equipment is located in the turbine building.

The turbine building is a steel column and beam structure with a reinforced concrete slab as ground floor (structural mat). The turbine generator is low-tuned by means of spring supports.

4.4.7.5 Other buildings

Auxiliary building.

The primary function of the auxiliary building is to provide protection and separation for the Seismic Category I mechanical and electrical equipment located outside the containment building. It also provides shielding for the radioactive equipment and piping that is housed within the building. A representative outline drawing of the auxiliary building is shown in Figure 4.4-8.

The auxiliary building is a Seismic Category I reinforced concrete structure which shares a common basemat with the containment building. The auxiliary building is a C-shaped section of the nuclear island that wraps around approximately 70 percent of the circumference of the shield building. Floor slabs and the structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

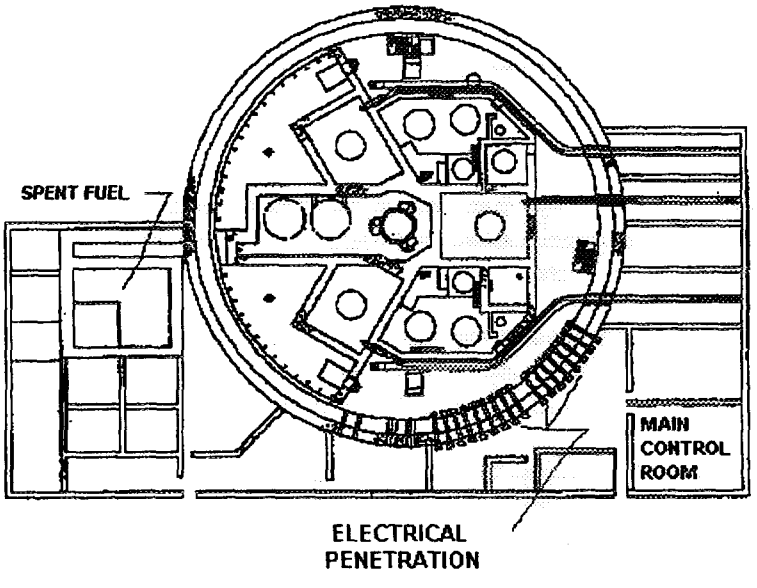


Figure 4.4-8 Nuclear island arrangement; plan view at operating floor elevation

The auxiliary building is located between the containment building and the turbine building and between the containment building and the annex buildings. Because of the configuration, the auxiliary building provides communication between the containment and the annex buildings.

The most significant equipment, systems, and functions contained within the auxiliary building are described in the following.

Fuel handling area: The primary function of the fuel handling area is to provide for the handling and storage of new and spent fuel. The fuel handling area provides the means for receiving, inspecting, and storing the new fuel assemblies. It also provides for safe storage of spent fuel.

The fuel handling area provides for transferring new fuel assemblies from the new fuel storage area to the containment building and for transferring spent fuel assemblies from the containment building to the spent fuel storage pool within the auxiliary building. The fuel handling area also provides the means for removing the spent fuel assemblies from the spent fuel storage pit and loading the assemblies into a shipping cask for transfer from the facility.

The release of airborne radiation in the fuel handling area, following any postulated design basis accident that could result in damage to the fuel assemblies or associated radioactive systems, does not result in unacceptable site boundary radiation levels.

Mechanical equipment areas: The mechanical equipment located in the radiological control areas of the auxiliary building are the spent fuel cooling system pumps and heat exchangers; the liquid and gaseous radwaste pumps, tanks, demineralizers and filters; the chemical and volume control pumps; and heating, ventilation, and air conditioning exhaust fans.

The mechanical equipment located in the clean areas of the auxiliary building are the heating, ventilating, and air conditioning air handling unit and associated equipment that service the main control room, the instrumentation and control cabinet rooms, and the battery rooms.

Containment penetration areas: The auxiliary building contains the containment penetration areas for mechanical, electrical, and instrumentation and control penetrations. The auxiliary building provides separation of the radioactive piping penetration areas from the non-radioactive penetration areas, and separation of the electrical and instrumentation and control penetration areas from the mechanical penetration areas. Also provided is separation of redundant channels/trains of mechanical and electrical equipment.

Main steam and feedwater isolation valve compartment: The main steam and feedwater isolation valve compartment is contained within the auxiliary building. The auxiliary building provides an adequate venting area for the main steam and feedwater isolation valve compartment in the event of a postulated leak in either a main steamline or feedwater line.

Radwaste building

The radwaste building is designed to house all the equipment needed in processing of solid waste for final disposal. In addition, the radwaste building is designed for the intermediate storage of all solid and solidified radioactive wastes from the plant.

As a design criteria for the minimum storage capacity, 5 years' normal operation has been used. The waste treatment processes and equipment have been designed to meet the European requirement of the maximum allowed solid waste disposal volume of 50 m^3 per year.

In the layout, as well as in the access and process control operations, the minimization of personnel exposures has effectively been taken into account.

The radwaste building is designed to consist of two main areas: the left hand side area (in the attached layout drawings) is dedicated for the spent resin processing, and the right hand side for processing of other types of wastes, mainly dry active wastes (DAWs).

Two alternative layouts for the radwaste building have been planned. The right hand side is equal in both of them. The left hand side, on the contrary, is different depending on the process chosen for the spent resin treatment.

Annex building

Annex building layout activities will be performed later in the development of the EPP programme.

Diesel generator building

The Diesel generator building houses the two non-safety-grade diesel generators and their associated auxiliary equipment.

4.4.8 Technical data

General plant data

Power plant output, gross	1 000	MWe
Power plant output, net		MWe
Reactor thermal output	2 910	MWt
Power plant efficiency, net		%
Cooling water temperature		°C

Nuclear steam supply system

Number of coolant loops	3 hot legs, 6 cold legs	
Primary circuit volume, including pressuriser	318.2	m^3
Steam flow rate at nominal conditions	1 590	kg/s
Feedwater flow rate at nominal conditions	1 590	kg/s
Steam temperature/pressure	283/6.7	°C/MPa
Feedwater temperature/pressure	221/7.2	°C/MPa

Reactor coolant system

Primary coolant flow rate	14 150	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	289.2	°C
Coolant outlet temperature, at RPV outlet	325	°C
Mean temperature rise across core	39.6	°C

Reactor core

Active core height	3.658	m
Equivalent core diameter	3.376	m
Heat transfer surface in the core	5 561	m^2
Fuel inventory	89.35	t U
Average linear heat rate	15.16	kW/m
Average fuel power density	32.57	kW/kg U
Average core power density (volumetric)	88.84	kW/l
Thermal heat flux, F_q	2.6	kW/m^2
Enthalpy rise, F_H	1.65	

Fuel material	Sintered UO_2	
Fuel assembly total length	4 800	mm
Rod array	square, 17×17	
Number of fuel assemblies	193	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	25	
Number of spacers	9	
Enrichment (range) of first core		Wt%
Enrichment of reload fuel at equilibrium core		Wt%
Operating cycle length (fuel cycle length)	18 or 24	months
Average discharge burnup of fuel	40 - 60 0 000	MWd/t
Cladding tube material	Zircaloy	
Cladding tube wall thickness0.57	mm	
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly	620	kg
Active length of fuel rods	3 658	mm
Burnable absorber, strategy/material	Wet annular burnable absorber, Integral fuel burnable absorber	
Number of control rods	57	RCCA
	16	Grey RCCA
Absorber rods per control assembly	24	
Absorber material	Ag-In-Cd	
Drive mechanism	Magnetic jack	
Positioning rate [in steps/min or mm/s]		
Soluble neutron absorber	boric acid	

Reactor pressure vessel

Cylindrical shell inner diameter	4 439	mm
Wall thickness of cylindrical shell		mm
Total height	12 046	mm
Base material: cylindrical shell	Carbon steel	
RPV head		
liner	Stainless steel	
Design pressure/temperature	17.24/343	MPa/°C
Transport weight (lower part)		t

RPV head	t	

Steam generators

Type	Delta 75, vertical, U-tube	
Number [Thermal capacity 970 MWt]	3	
Heat transfer surface	6 986	m^2
Number of heat exchanger tubes	6 307	
Tube dimensions [ID/OD]	15.42/17.46	mm
Maximum outer diameter	4 500	mm
Total height	21 000	mm
Transport weight	480	t
Shell and tube sheet material	Carbon steel	
Tube material	Inconel 690 TT	

Reactor coolant pump

Type	Single-stage, centrifugal pump with canned motor	
Number	6	
Design pressure/temperature		MPa/°C
Design flow rate (at operating conditions)	2 358	kg/s
Pump head	73	
Power demand at coupling, cold/hot	2 240*)	kW
Pump casing material		
Pump speed	1 450*)	rpm

Pressuriser

Total volume	59.5	m^3
Steam volume: full power/zero power	22.7	m^3
Design pressure/temperature	17.2/	MPa/°C
Heating power of the heater rods	1 884	kW
Number of heater rods		
Inner diameter		mm
Total height		mm
Material		
Transport weight	105	t

*) approximate value

Pressuriser relief tank (if any)

Not applicable

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Dry, free-standing, steel	
Overall form (cyl.)	cylindrical	
Dimensions (diameter/height)	46/67.8	m
Free volume	82 000	m^3
Design pressure/temperature (DBEs)	0.344/140*)	kPa/°C
(severe accident situations)	t.b.d.	kPa/°C
Design leakage rate	0.12	vol%/day
Is secondary containment provided?	yes, in concrete	

Reactor auxiliary systems

Reactor water cleanup,	capacity	6.2	kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage	kV
	rated capacity	MVA
Plant transformers,	rated voltage	kV
	rated capacity	MVA
Start-up transformer	rated voltage	kV
	rated capacity	MVA
Medium voltage busbars (6 kV or 10 kV)		

*) approximate value

Number of low voltage busbar systems		
Standby diesel generating units: number	2	
rated power	6	MW
Number of diesel-backed busbar systems	2	
Voltage level of these		V ac
Number of DC distributions		
Voltage level of these		V
dc		
Number of battery-backed busbar systems		
Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	TC4F-52	
Number of turbine sections per unit (e.g. HP/LP/LP)		
Turbine speed	1 500	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	6.65/282	MPa/°C

Generator

Type	3-phase, synchronous	
Rated power		MVA
Active power		MW
Voltage	22	kV
Frequency	50	Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type		
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate		m^3/s
Cooling water temperature		°C
Condenser pressure		hPa

Condensate pumps

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

Condensate clean-up system

Full flow/part flow	
Filter type	

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/°C

Feedwater pumps

Number	
Flow rate	kg/s
Pump head	
Feedwater temperature	°C
Pump speed	rpm

Condensate and feedwater heaters

Number of heating stages	
Redundancies	

4.4.9 Project status and planned schedule

The Phase 1 of the European Passive Plant Evaluation Programme has been completed in June 1996.

The results of the Phase 1 programme has been summarized in two major reports.

The first is the Conceptual Design Study Report. This report focuses on the results of the design and evaluation efforts made during Phase 1 and includes both the conceptual design and analyses/evaluations performed during Phase 1 and the final version of the system specification documents.

The second is the Preliminary Safety Case Report. The body of this report summarizes the conceptual designs developed during Phase 1 and also summarizes the results and conclusions of the design evaluation and analyses that were performed to support the systems design.

The ultimate objective of Phase 2 of the programme is to develop design details and perform supporting analyses to produce a Standard Safety Analysis Report (SSAR) for submittal to European Safety Authorities. The first part of Phase 2, "Design Definition" will focus on the definition and design of important systems and structures. The second part, "Verification of Design and SSAR Preparation", includes both the analyses and evaluations required to demonstrate the adequacy of the design, and the preparation of SSAR.

The first part of Phase 2 of the programme will start at the beginning of 1997 and will be completed at the end of 1998.

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4.5 EPR, NPI, FRANCE/GERMANY

4.5.1 Introduction

The EPR (the European Pressurized Water Reactor) is the designation for a development effort by Nuclear Power International and its parent companies, Framatome and Siemens, in co-operation with Electricité de France and German Utilities, aiming at achieving a new improved nuclear power plant design that will become an acceptable and attractive alternative for meeting energy demands in the future.

The conceptual design features of the EPR were harmonized between the partners by the end of September 1993 and submitted to the DFD (Deutsch - Franzoesischer Direktionsausschuss), a joint committee of the French and German safety authorities, for assessment. The response of the DFD to certain safety features was received by end of 1994/beginning of 1995. The early feedback from the French and German safety authorities was of vital importance for the continuation of the development work and a precondition for starting the basic design.

NPI can rely on the huge experience gained by its parent companies; they have by now built or have under construction more than 100 nuclear power plants in various countries throughout the world. The total installed capacity exceeds 100 000 MW - about 25% of the total world-wide figure, and more than any other manufacturer.

The outstanding Franco-German experience background of more than 1200 reactor years of operation made it natural to decide that the design of the new generation of pressurized water reactors should build on the ever-growing experience background from this fleet of operating plants. The evolutionary way of development has proved very successful in both countries since the beginning of their nuclear programmes, and therefore a continuation of this development process was found preferable. The feedback of operating experience and the results of in-depth studies performed for the existing units will, beyond doubt, make it possible to attain a significant increase in reliability, safety and economy, even though already starting from a high level.

Following the conceptual design phase of the so-called Common Product conducted by NPI, Framatome and Siemens, from 1989 through 1991, Electricité de France (EdF) and several major German utilities decided to merge their own development programmes, - the N4 Plus and REP 2000 projects on the French side and the further development of the KONVOI technology on the German side, - with the NPI project. From that time on, the NPI project became one single common development line for both countries, named EPR (european pressurized water reactor). The utilities decided at that time to establish, together with other european utilities, specifications that would represent common utility views on the design and performance of future nuclear power plants. A first draft of the generic (non-design-specific) sections of these European Utility Requirements (EURs) was issued in April 1994, and an extended version B taking into account comments to the draft from utilities and vendors was issued in April 1996.

The following description of the EPR is based on the harmonization between the partners, and it may be subject to future modifications following the assessment by the safety authorities.

4.5.2 Description of the nuclear systems

4.5.2.1 Primary circuit and its main characteristics

The primary loop configuration is the same as that of existing designs and can be considered well-proven.

The sizing of the reactor pressure vessel (RPV), steam generator (SG) (especially secondary side) and pressurizer (PZR) incorporates increases of the respective volumes compared to the current designs.

In the RPV design, the free water volume between the level of the reactor coolant lines and the top of the active core is increased in order to improve the mitigation of LOCA (smaller breaks) by prolonging the period until beginning of core uncovery in case of LOCA or minimizing the core uncovery depth if any.

At the same time the increase of this volume contributes to an improvement in the mitigation of accidents at shutdown conditions (in particular mid-loop operation), e. g. with loss of the Residual heat removal system (RHR), by providing longer grace periods.

For the pressurizer a large volume both in terms of water and steam phase is provided in order to smoothen plant response to relevant operating transients and accidents.

The large water volume of the SG secondary side supports smoothing of normal operating transients and reduces the potential for unplanned reactor trips. In case of a total loss of all feedwater supply (incl. emergency feedwater), the postulated dryout time of the SG will be > 30 min.

The valve configuration of the primary side overpressure protection aims at avoiding the response of "non-isolatable valves" in plant conditions with a potential for radioactivity release. The valves are mounted to the top of the pressurizer in order to minimize the length of high pressure piping. Each discharge train is provided with two safety valves in series; this tandem arrangement makes it possible to isolate a stuck-open safety valve without decreasing overpressure protection capability. Automatic opening of the main valves (of own medium operated type) is actuated by pilot actuators dedicated to each individual safety valve. During normal operation the valve assigned to the discharge function is closed, the valve assigned to the isolation function is open. For operation at lower temperatures, during stretch-out operation, the pilot actuators of at least one train are provided with remotely adjustable setpoints.

With the chosen arrangement of pressurizer discharge, the following safety functions can be performed:

- over-pressure protection of the reactor cooling system by automatically initiated discharge of either steam, water or two-phase fluid,
- depressurization of the reactor cooling system by discharge of steam, water or two-phase fluid in plant conditions when pressurizer spraying is not available or not efficient,
- discharge of the reactor cooling system to enable residual heat removal in case of unavailability of the secondary side heat removal (feed and bleed),
- discharge of the reactor cooling system in a postulated core melt situation to guarantee depressurization to a sufficiently low level that would rule out the high pressure core melt accident and its severe consequences.

On the secondary side of the SGs, three discharge trains will be arranged on each main steam line at the outlet of the containment. The discharge trains on each line will be arranged as follows:

- One discharge line will be equipped with a relief valve (normally open) and an isolation valve (normally closed).
- The two other discharge lines will be equipped with non-isolatable safety valves.

In the overall concept of secondary side pressure limitation and heat removal the relief valve represents the second line of defence-in-depth (beyond the turbine bypass actuation level forming the first line) and assures safety-grade controlled heat removal and pressure limitation. The safety valves are dedicated to the final level of defence in terms of pressure limitation.

4.5.2.2 Reactor core and fuel design

The evolutionary approach chosen for the development of the EPR has led to a core design which is firmly based on the experience gained with existing plants. The core is built up by 241 mechanically identically designed fuel assemblies; somewhat more than in currently operating units. Each fuel assembly consists of 264 fuel rods and 25 guide tubes arranged in a 17x17 array; their active length is 4.20 m.

The fuel rods are made of Zircaloy tubing containing uranium dioxide ceramic pellets, of which the initial enrichment is below or equal to 5.0 Wt%.

The average linear heat generation rate is about 15.5 kW/m, giving prospect of achieving average batch burnups of up to 60 GWd/tU. The design offers a high degree of flexibility with respect to cycle length adaptations, allowing fuel cycle cost reductions by high burnups and low leakage loading patterns.

Basic safety objectives are met by designing the core to have stabilizing reactivity coefficients under all operation conditions. Reactivity control is accomplished by changing the boron concentration in the primary coolant and by moving control assemblies. As a rule, slow reactivity changes caused by changes of xenon concentration and burnup are compensated by changes of the boron concentration, while fast reactivity changes for adaptation of the power level are compensated by control rod insertion or withdrawal.

The core is designed for UO_2 fuel elements application and incorporates the capability also to insert MOX-fuel assemblies up to about 50%.

Some fuel assemblies contain burnable absorber (Gd_2O_3) to suppress high excess reactivity, especially in the first core. The reactor power level is monitored by the ex-core instrumentation. The capability to predict and to measure the three dimensional power distribution in the core is the duty of the in-core instrumentation system which consists of the aeroball system and the self-powered detector system. The in-core instrumentation penetrates the reactor pressure vessel head from the top at only very few positions. A cross-section of the core, showing the location of the core instrumentation, is depicted in Figure 4.5-1.

4.5.2.3 Fuel handling and transfer systems

The burnt out fuel assemblies are transferred to the fuel pool located in the fuel building. The new fuel assemblies are stored in a storage place in the fuel building to enable easy access thereto. The number of fuel assemblies stored are sufficient for a reload, 102 positions are available. Inside the reactor building, a loading machine transfers the burnt out as well as the new fuel assemblies into or out of the reactor. An opening in the reactor building wall enables the transfer between the two buildings.

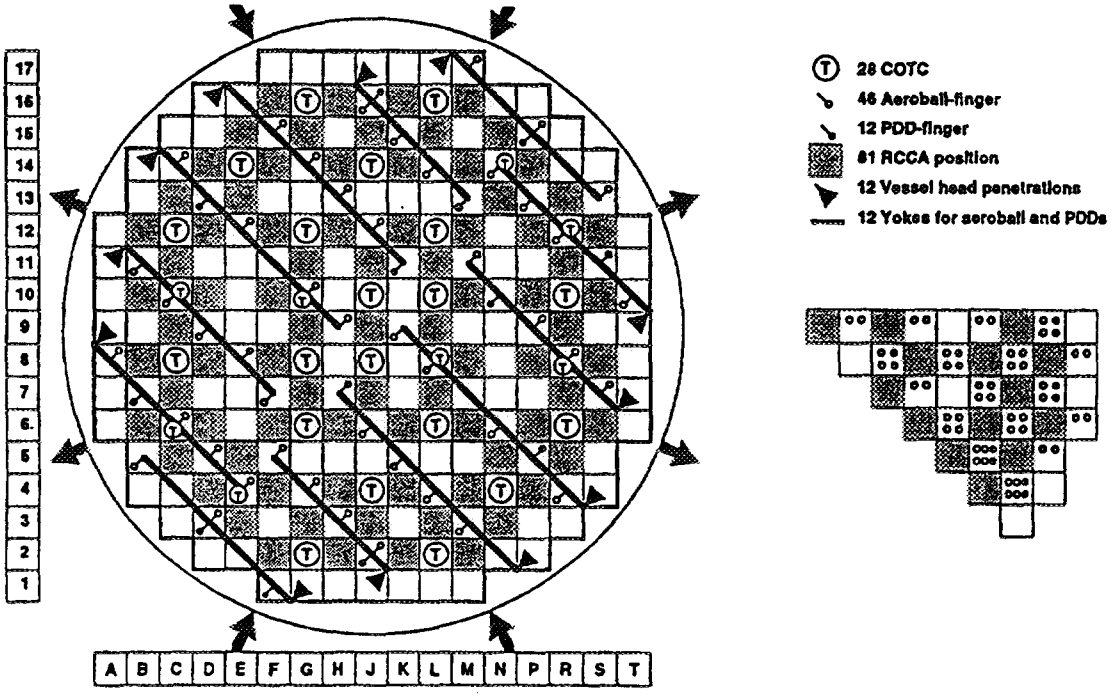


Figure 4.5-1 EPR - Core instrumentation

4.5.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel is designed for a life time of 60 years, not exceeding a total neutron fluence of 10^{19} nvt. This is achieved by provision of a rather large water gap and a heavy reflector around the core.

The upper part of the reactor pressure vessel will be machined out of one single forging. The flange is made as an integral part of the nozzle shell. The nozzles itself are of the on-set type so that non-destructive examination can be easily performed from the inside. The nozzles are located as high as practicable above the core upper edge to increase the hydrostatic pressure for reflooding and to avoid the loop seal effect.

The reactor pressure vessel is internally clad with two layers of low carbon stainless steel. The inside surface is ground after cladding to a finish as required for ultrasonic inspection. Generally, in-service inspection is performed from inside the vessel; an access from the outside, between the outer wall and the thermal insulation, is also provided, however. Inspection from the outside will be performed, if the internal inspection should reveal indications which cannot be adequately characterized from the inside.

Steam generators

The steam generators feature an axial economizer to provide a steam pressure increase of about 3 bar when compared to a boiler type of the same heating surface.

The material for the heating tubes will be chosen from either Incoloy 800 or Inconel 690. Both materials have proven excellent properties regarding corrosion resistance and are exchangeable without affecting steam generator design parameters. The heating tubes are supported by perforated plates.

With respect to the pressure boundary, the same material as for the reactor pressure vessel will be chosen.

Pressurizer

The pressurizer is of conventional design but with an enlarged free volume. The spray systems for normal operation and auxiliary spray are completely separated from each other.

In addition, in order to accommodate a life time of 60 years and meet the plant power flexibility requirements, several separate spray lines are provided and operated alternatively.

The spray lines are welded through a blind cover and equipped with a spray nozzle each. This design is easy to dismantle, inspect and replace. The spray system delivers a permanent flow to the spray nozzles to minimize thermal transients upon fast valve opening.

The heaters are flanged to the penetrations in order to be easily replaced and inspected.

All pressure boundary parts, except for the heater penetrations, are made of ferritic steel grade; basically the same as used for the reactor pressure vessel. The penetrations are in stainless steel and welded with an Inconel material.

Reactor coolant pumps

The reactor coolant pumps are of well-proven design, as already used in plants in France and Germany. The reactor coolant pumps are provided with a standstill seal in order to assure leak-tightness of the shaft seal without the need of an active seal water supply system under conditions when the pump is at rest, e.g. in the event of a station blackout.

Main coolant lines

With respect to the material of the main coolant lines, two options are still under consideration; either forged ferritic steel with austenitic cladding or forged stainless steel will be used. In any case, the break

preclusion concept will be applied. A high quality in design, construction and surveillance enables preclusion of the current pipe break assumptions; consequently a catastrophic failure of a main coolant line is ruled out as regards its possible mechanical effects. However, a mass flow equivalent to a double area break of a main coolant line is still assumed for the design of e.g. the emergency core cooling system.

Reactor internals

The core barrel flange rests on a ledge machined from the flange of the reactor pressure vessel and is preloaded by an elastic system. The fuel assemblies are placed directly on a flat perforated plate, machined from a forging of stainless steel and welded all around to the core barrel. The fuel assemblies are centered by two pins each, which are screwed to the core plate.

The cooling water flows through the core plate through four holes dedicated to each fuel assembly. These holes can be calibrated in such a way that a flat flow profile is achieved.

The space between the polygonal outside shape of the core and the cylindrical inner surface of the core barrel is filled by a stainless steel structure to reduce the fast neutron leakage and to flatten the power distribution. This structure is called the heavy reflector and represents an innovative feature compared to earlier designs, aiming at savings concerning the enrichment requirements.

The bulk of the internals is made of low carbon stainless steel in line with the current practice to prevent intergranular stress corrosion cracking in primary water environment.

4.5.2.5 Reactor auxiliary systems

Chemical and volume control system

The configuration of the chemical and volume control system is outlined in Figure 4.5-2.

In the EPR, the system has beside operational functions also safety-related (with respect to risk reduction category A measures) functions. The safety-related functions aim at providing make-up to the reactor coolant system in the event of loss of inventory due to limited leakages and providing boration capacity up to a concentration as required for cold shutdown under operational and certain accident conditions.

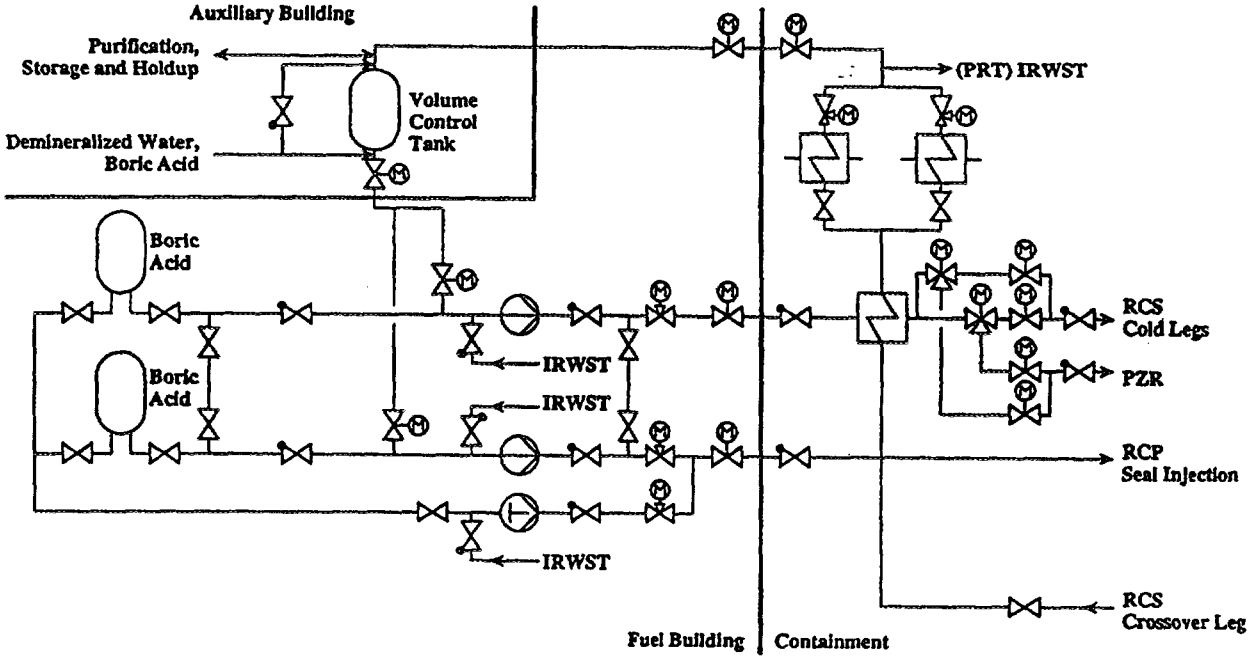


Figure 4.5-2 EPR - Chemical and volume control system configuration

In addition, the system provides a means for pressure control of the primary circuit by supplying auxiliary spray to the pressurizer, thereby condensing steam and reducing primary system pressure, and seal water to the reactor coolant pumps.

The operational functions of the system are in summary: to control the water inventory, the water quality and the boron concentration in the primary system. Additionally, the system adjusts the chemical composition of the reactor coolant system and removes dissolved gases by degasification of the letdown flow.

The main function of the chemical and volume control system is realized by a combination of charging and letdown flows. Adjustment of the letdown flow permits adjustment of the water inventory of the primary system, while the adjustment of boron concentration in the makeup flow provides reactivity control. The letdown flow is taken from one of the cross-over legs of the reactor coolant system and subsequently cooled by a regenerative heat exchanger and a high pressure cooler. After this cooler, the pressure is reduced to below high-energy conditions and routed to the fuel building where it is further processed, e.g. purified. Letdown to the In-containment refuelling water storage tank is provided via a bypass line to allow letdown even when parts of the chemical and volume control system are not available.

After processing, the coolant flow is directed to the suction side of the high pressure charging pumps. Injection of the charging flow is ensured by one of three charging pumps, two of which are high pressure centrifugal pumps, - the third being a positive displacement pump, - thus providing diverse and redundant means for reactor coolant system boration and seal water supply. From the charging pumps, the flow is directed via the regenerative heat exchanger to the reactor coolant system. A second injection line is provided directly to the reactor coolant pump seal injection lines.

Further reactor auxiliary systems

The following auxiliary systems (1-4) and safety-grade systems (5-7):

1) - Coolant treatment system
2) - Degassification system
3) - Waste water treatment system
4) - Waste conditioning system
5) - Fuel pool cooling system
6) - Component cooling water system
7) - Service water system

are under development. System descriptions have to be added later.

4.5.2.6. Operating characteristics

The EPR is a nuclear island for a power output of about 1500 MW. The primary components are enlarged relative to current types, the safety and operating systems have been redesigned or updated to the conditions required.

The EPR is designed for being operated between 20 and 100% of rated generator power. In the power range between 50% and 100% load, the control systems will keep the average coolant temperature constant in accordance with the so-called part load diagram; the main steam pressure will vary between 8.4 and 7.25 MPa. In the lower power range, below 50% power, the main steam pressure is kept constant at 8.4 MPa and then the average coolant temperature will vary with the load. The advantage of this control strategy is that it results in the lowest demands on the chemical and volume control system, and loads on the pressuriser surge line and the control rod drive mechanisms during load changes in the most frequent operation mode.

In the lower power range, a continuation of the constant average coolant temperature mode would penalise the secondary side design (by a design pressure increase) and eventually also the primary side design pressure.

The control and operational systems are designed to provide the EPR with a high capability to follow the actual power demands of the grid. The load follow capability can briefly be summarised as:

- ± 5%/min ramp load change within 50 and 100% of rated thermal power (±2.5%/min within the 25 to 50 % range);
- ± 10% step load change within 20 and 100% of rated power;
- + 20% power increase within 2 minutes;
- 100-25-100% load follow operation cycles, with several load changes per day; and
- primary and secondary grid frequency control equivalent to ~ ±10%.

The load changes can either be initiated by the operator or completely remotely controlled. Important plant parameters are maintained within operational ranges automatically by control system functions, and the setpoints for the main NSSS controls are adjusted automatically; all plant parameters remain far from the triggering setpoints of any safety system during normal plant operation.

In addition, the EPR is designed to withstand without tripping of the reactor, events like: turbine trip, full load rejection, trip of one feedwater pump, and malfunction of a single control system.

4.5.3 Description of turbine generator plant system

The EPR project is basically restricted to the nuclear steam supply system (NSSS) design, intended to be a common design for France and Germany. With respect to the conventional parts of the plant, the decision on global as well as detailed architecture has been left to the utilities. The description of the turbine plant in this report refers to an EPR to be built in France.

The strategy adopted for the turbine generator and its associated systems is to improve plant efficiency by utilization of high-performance components; the extra costs associated with high-performance components would be offset by the gain in efficiency.

4.5.3.1 Turbine generator plant

The turbine of the EPR is a new development based on the Arabelle turbine of the N4. The nominal power output is 1500 MWe. The saturated steam coming from the steam generators is flowing to the inlet valves of the high medium pressure (HMP) cylinder of the turbine via four steam lines. No header is provided beteen the steam lines and the HMP cylinder. After expansion in the HMP cylinder, the steam is routed through two steam moisture separator and reheater units, located on the sides of the HMP cylinder, and then to the three low-pressure cylinders in parallel.

After expansion in the low-pressure cylinders, the steam goes to the condenser which is of modular type for easy maintenance, with two modules for each low-pressure cylinder. Depending on the site characteristics, the tube material will be stainless steel for river sites or titanium for coastal sites.

The condenser cooling water system is a two-train system, each equipped with a low-speed motor-driven pump capable of supplying 50% of the nominal cooling flow. The pump casing is made of concrete. In case of pump malfunction, the turbine is not tripped, and the power output is decresed to match the remaining flow.

The turbine bypass to the condenser is designed to accept 50%of the rated steam flow to the turbine.

The generator is a two-pole type, using hydrogen as the rotor coolant and water for cooling of the stator windings. The rotor is directly coupled to the turbine. The generator rated power is 1900 MVA and the cos φ is equal to 0.85.

4.5.3.2 Condensate and feedwater systems

Water coming from the condenser is pumped through four low-pressure heaters to the deaerator and the storage tank by means of two 50% condensate pumps. Three high-pressure electric motor-driven

pumps, each with a capability of providing 50% of the rated flow (one pump is in standby) circulate the feedwater through the high-pressure heaters for injection in the steam generators.

4.5.3.3 Auxiliary systems

The most important auxiliary systems associated with the conventional part of the plant are:

- the auxiliary steam system, providing steam generated by two electrical heaters to the seals installed on the low-pressure cylinders;
- the circulating water system, providing cooling water to the condenser, and whose pumps are located in the turbine building;
- the chemical additive system, of which the main function is to keep the pH of the water above 9, to prevent corrosion acidic media;
- the steam generator drain system: the steam out of the flash tank is re-injected in the feedwater tank; and
- the turbine drain system: steam drained from the turbine cylinders is circulated through high and low-pressure heaters for feedwater heatup and is re-injected in the feedwater system upstream the high-pressure heaters.

4.5.4 Instrumentation and control systems

4.5.4.1 I & C Design concept, including control rooms

I&C structure

The functional requirements and failure models governing the design of the I&C systems are based on the overall safety criteria for system design and the functional requirements on the process systems. This implies a requirement for independent I&C subsystems in order to ensure that a loss of one subsystem will influence the remaining I&C systems only marginally.

The I&C systems and equipment are divided up into three classes (IE1, IE2, IE3) in accordance with their importance for safety and with respect to required reliability, performance, failure behaviour, maintenance, testing, and QA; in addition, there is a non-classified category (NC). The main features of these categories are:

- The I&C of the IE1 class that is used for automatic actuation and control of safety-grade systems, shall withstand a single failure also during periods of maintenance and testing. In addition, there must be no spurious actuation during maintenance or testing in case of a failure occurring anywhere upstream the last voter. This leads to a need for a fourfold redundant and divisionally separated structure.
- The I&C of the IE2 class, including information means and means for manual actions for mitigating measures during an accident, has a functional structure that can cope with a single failure - without the additional requirements related to maintenance and testing. Special emphasis is laid on the qualification of the back-up control and information means (safety control area), including the related software.
- The I&C of the IE3 and NC classes, that is used for normal operation of the plant, including the control of major plant parameters, limiting conditions of operation (LCOs), information and operation and other non-safety I&C functions, is specified in a case by case approach.

During the basic design stage, it will be ascertained that different I&C functions are assigned to the proper class, and that any I&C function with a higher classification will have priority over lower classified (less important) ones. Failures in a lower classified function are not allowed to jeopardize functions of a higher classification.

Applied technology

The proposed I&C automation and Man-Machine-Interface systems are based on utilisation of digital technology, preferably with "off-the-shelf" electronic components.

The potential for common cause failures is reduced by consequent use of functional diversity, for initiating parameters and actuation channels, and by distributing diverse I&C functions to I&C systems. Further, a formal specification of the I&C systems reduces failure potentials of software specification and makes the software easily verifiable.

Safety I&C (IE1, IE2)

The safety I&C functions shall have a high reliability so that they will not be a dominant contributor to the unavailability of safety systems.

The safety I&C is of redundant architecture, and designed to limit the consequences of equipment failures or malfunctions that may result from failure inducing events within I&C systems (single failure) and their consequential effects (with active or passive failure mode). Connections between redundant trains are necessary for exchange of information and commands, but they must not impair the independence. To this end, the redundant trains, or divisions, are installed with physical separation and with a minimum number of interconnections.

Interconnections are energetically decoupled against overvoltages from a disturbed division (e.g. by means of fibre optics), and erroneous signals from a disturbed division are prevented from affecting the other divisions by means of majority voting or signal coding. Necessary safety actions must be performed from the undisturbed divisions independent of the state of a disturbed division.

Appropriate measures are provided to cope with common cause failures (CCF) in order to meet the overall probabilistic design targets. CCFs and their consequences can result from different sources (e.g. faulty manufacturing, erroneous design, equipment failures, and environmental conditions during the course of accidents). CCFs are unavoidable, but the probability of their occurrence must be kept at a low level. Special probability values for digital I&C are practically nonexisting, and evaluations must therefore be based on engineering judgement.

The safety I&C functions, systems and associated equipment shall not be jeopardized by the operational I&C, and they are therefore decoupled from the operational I&C when interconnections can not be avoided. To this end, a "priority control" strategy is implemented; a safety command to an object used by both safety and operational I&C overrides any non-safety command.

Operational I&C

The operational I&C encompasses all I&C functions and associated systems and equipment for normal operation. It contains the measurements, the signal conditioning, open- and closed-loop controls, the signal processing and the data transfer to the man-machine interface.

The tasks of the open- and closed-loop controls are mainly to control the nuclear power generation during normal operation and anticipated operational occurrences in such a way that predetermined setpoints for relevant process variables are reached and maintained, to protect all mechanical equipment with high investment costs via redundant equipment protection, and to provide information for displaying the plant status for normal, upset and accident conditions and for documentation of all relevant process data.

Operational controls are operating in such a way that a sufficient margin to the actuation values of the safety I&C is maintained.

Man-machine interface facilities, and control rooms

The general structure of the man-machine interface facilities and control rooms of the EPR is depicted on Figure 4.5-3.

The main control room (MCR) is a screen based control room with an overview panel. It is used for process control during normal, or accident situations including outages. In addition, the MCR has a safety control area with back-up control means. Further functions that are ensured from the main control room or from adjacent rooms are security surveillance, fire protection monitoring, radiation monitoring, management

of maintenance and periodic testing, external and internal communication, access to documentation and to recorded information.

The MCR contains 3 operator work positions (all of the same design) which are used for process control in all plant conditions via operational I&C. A shift supervisor console offers operational and safety-qualified information to the shift supervisor, and/or to the safety engineer. It is equipped with communication means and space for administration work. The operator work positions are dedicated to the operators of the primary and secondary loops, and to the auxiliary operation or back-up purposes.

A plant overview panel is visible from all work places and will be used for the co-ordination among the operators and for the transfer between normal and back-up means.

The safety control area (with the back-up control means) in the MCR is used in the event of major losses of the normal control means. It can be used for the safe shutdown (hot or cold) of the plant or to perform post-accident operation. This area could also be screen based. The area constitutes a safety-relevant man-machine interface, and the related equipment is qualified accordingly.

The EPR is provided with a remote shutdown station (RSS) which is designed for transferring the plant to and maintaining it in safe shutdown conditions, in case of unavailability of the main control room without loss of operational or safety I&C systems. The RSS is equipped with internal and external communication means.

4.5.4.2 Reactor protection and other safety systems

The reactor protection system and other safety I&C systems are digital systems. They are characterized by divisional separation of their independent subsystems and the use of voting functions (2/4, 2/3, ..) for protective actions. The structure meets the requirements of the safety I&C functions described above.

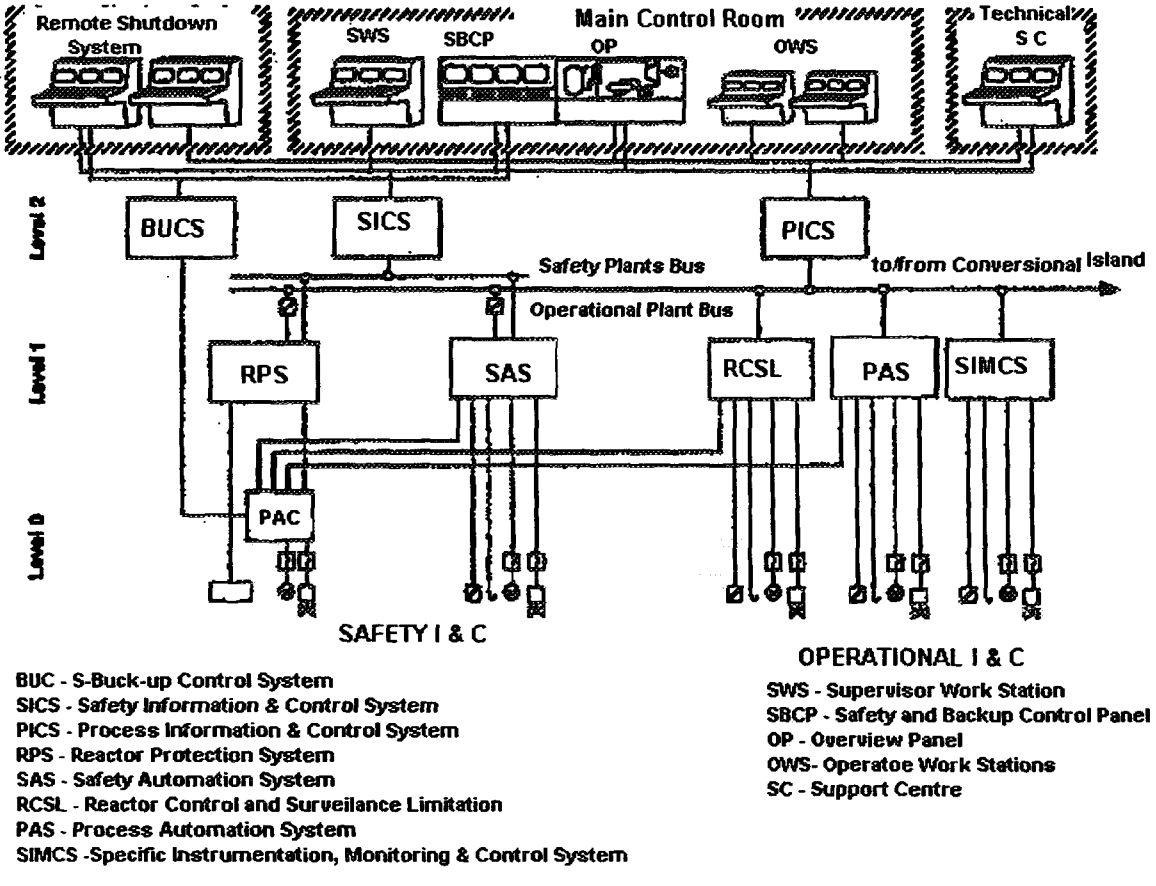


Figure 4.5-3 EPR - Control systems structure

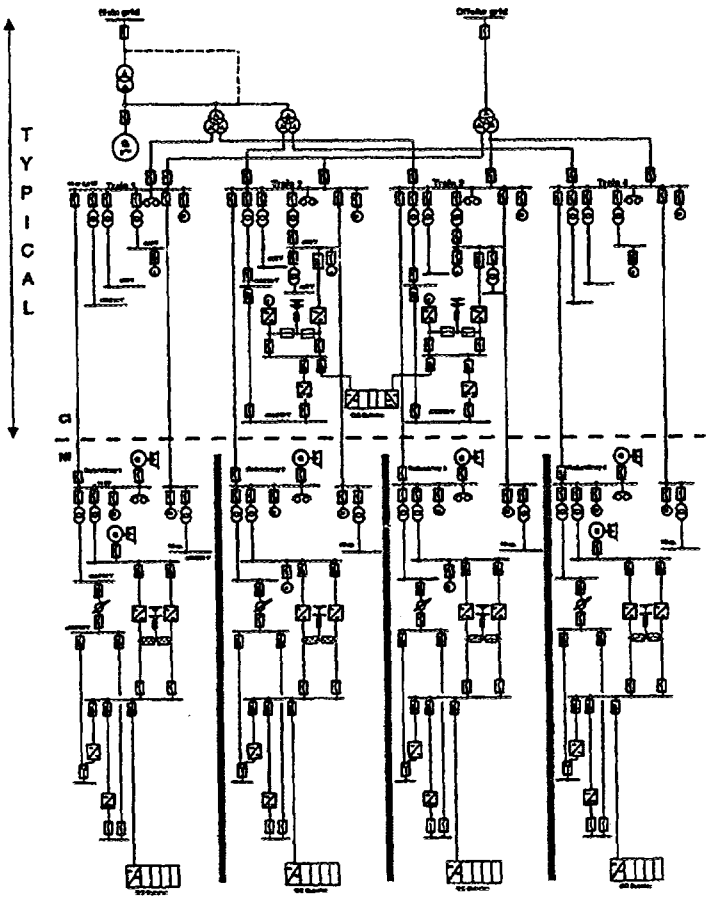


Figure 4.5-4 EPR - Electrical power systems, single line diagram

4.5.5 Electrical systems

The basic arrangement of the electrical systems is shown in the single line diagram in Figure 4.5-4.

The main features are:

- Power delivery to the 400 kV grid via generator breaker (HV) and step-up transformer;
- Two three-winding step-down transformers for power supply during startup and load operation;
- One three-winding auxiliary transformer for power supply from a second offsite grid, for plant shutdown;
- A four-train arrangement for the operational power supply (in at least 2 divisions);
- A four train-four division concept for the safety-grade power supply systems, in accordance with the process systems design;
- Four Diesel generator sets in separated buildings, for the emergency power supply;
- Diversity of emergency power generation sources by two additional small diesel generators connected at 690 V level;
- Four battery sets (220 V DC), capacity for 2 hours at full load
- I&C power supply via DC/DC converters or directly from batteries at the suitable voltage
- AC voltage levels of 11 or 6.6 kV, and 690 and 400 V
- Dry-type low voltage transformers, directly connected to the low voltage busbars.

4.5.5.1 Operational power supply systems

The operational power supply systems are, from an electrical point of view, arranged in four trains, with four MV busbar systems which feed the largest motor loads, the four safety-related MV busbar

systems, four LV busbar systems on the conventional island (CI) and on the nuclear island (NI), and two trains of DC systems and uninterrupted AC systems (on Trains 2 and 3).

In order to improve the power supply availability, the two operational DC systems are provided with additional rectifiers which are connected to LV busbars of the two other CI trains (Trains 1 and 4); in addition, there are also "back-up" feeders from two 690 V CI busbars of Trains 2 and 3 to the corresponding two safety-related 690 V NI busbars that feed the safety-related DC systems.

4.5.5.2. Safety-related systems

As noted above, the safety-grade systems are arranged in four trains that are physically separated, each train containing a MV busbar, a number of LV busbars, an emergency diesel generator, a DC power supply system and an uninterrupted AC power supply system.

The safety loads and some non-safety loads which require emergency power supply are connected to the Emergency power supply system. The safety loads correspond to the items which are necessary to shut down the reactor safely, keep it in shutdown condition, remove the residual heat and the stored heat and to prevent impermissible release of radioactive substances. Other loads e.g. emergency exit lighting are also connected to this system. Depending on the required power, the loads are distributed to the different voltage levels.

The Emergency power supply system consists of four redundant and physically separated trains, each with a diesel generator of approximately 5 MW and a set of 220 V batteries with a capacity for 2 hours operation. The time period of unavailability of the grid has been assumed to be 24 hours. In the case of failure of the emergency power supply system, two smaller diesel generator sets with a capacity of 1-1.5 MW each and another battery set, also for 2 hours operating time, will take over.

DC power supplies are required for the function of I&C equipment (possibly via DC/DC converters) and protective devices, as control voltage for circuit breakers and for the DC/AC inverters that ensure an uninterrupted AC power supply. The DC power supply system is also divided into four separated redundant trains. A battery and two chargers are assigned to each redundant train. The batteries are designed with a capacity that will be sufficient to cover 2 hours at design load.

An uninterrupted power supply of 400/230 V AC is needed for the event of a failure of the operational power supply system. It will be established by means of DC/AC inverters, one per train. Each of the inverters is fed by the DC system which is assigned to the train. In the event of a failure of one inverter, the corresponding voltage distribution board is automatically switched over to the emergency diesel power system of that train.

4.5.6 Safety concept

4.5.6.1. Safety requirements and design philosophy

The strategy pursued for the EPR is to further enhance the already very high safety level attained at French and German plants. This strategy implies improving the prevention of accidents, including severe accidents, and adding features, mainly related to the containment, to mitigate the consequences of postulated severe accident scenarios - including core melt situations - to avoid need for stringent off-site countermeasures. The probability of such postulated accidents has been significantly reduced.

The plant design is based on a deterministic approach and considers "Risk Reduction" measures.

Deterministic design basis

In the deterministic analysis the different events are categorized in four Plant condition categories (PCCs) in accordance with their anticipated frequency of occurrence; PCC1 covers normal operation states, and PCC2 to PCC4 envelop transients, disturbed states and accidents.

Stringent radiological limits are applied for normal operation and anticipated operational occurrences as well as for accidents.

Risk reduction

The EPR design takes into consideration also events beyond the Deterministic design basis, - events with multiple failures and coincident occurrences up to the total loss of safety-grade systems, - on a probabilistic basis in order to quantify the residual risk. Severe accident design release limits have been specified on the basis that no stringent off-site emergency response actions (such as evacuation or resettlement) would be needed outside the immediate vicinity of the plant.

The overall safety objectives that have been set for the EPR, require that:

- the probability of core damage (CDF) shall be below 10^{-5}/year; and
- the probability of large releases to the environment shall be below 10^{-6}/year,

including all events and all reactor operating states.

In order to meet these objectives, some specific probabilistic design targets have been defined for the design phases:

- For internal events at power operation, the CDF shall be $< 10^{-6}$ per year;
- The contribution from shut-down states to the CDF for internal events shall be less than from the power states; and
- The CDF for internal events associated with early loss of containment function shall be below 10^{-7} per year.

Two Risk reduction categories (RRCs) have been introduced, and representative scenarios defined for both; - RRC-A relates to additional features to prevent accidents from progressing to a core melt situation, and RRC-B to prevention of large releases, - in order to provide a design basis for risk reduction features. Typical examples on risk reduction features are:

- primary system discharge to the in-containment refuelling water storage tank, in the event of total loss of secondary side cooling (RRC-A); and
- features for spreading and cooling of corium, for hydrogen recombination, and for containment heat removal in the event of a core melt situation (RRC-B).

The safety assessment of RRC-A features is performed in the form of a level 1 PSA, whereas the assessment of RRC-B features for considerable time will be largely deterministic, since a reliable level 2 PSA will only be possible at the end of the EPR design.

External and internal hazards

External and internal hazards are normally not assigned directly to specific plant condition categories or risk reduction categories, in order to avoid the study of numerous sequences. But the main principles behind the deterministic design basis and the risk reduction approach (namely: the more probable the event, the more conservative rules and acceptance criteria) are applied also for dealing with external and internal hazards.

External hazards are to a certain extent site-dependent. The possibility of chosing the boundary conditions in such a way, that it should be possible to construct the EPR on most potential sites is being considered. Sites with an extreme external hazard potential are not taken into consideration as potential sites.

4.5.6.2 Safety systems and features (active and passive)

Safety systems configuration

Important safety systems (safety injection, emergency feedwater, main steam relief, cooling chain, emergency electric power) are arranged in a four train configuration as depicted in Figure 4.5-5.

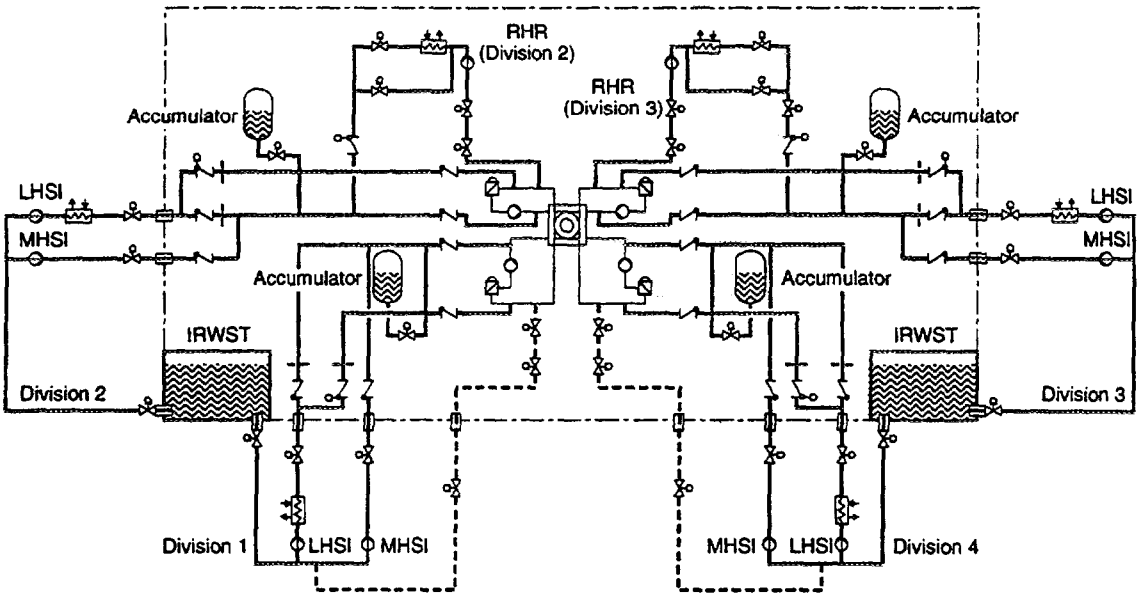


Figure 4.5-5 EPR - Configuration of fluid safety systems

The layout comprises four separate divisions, corresponding to the four trains. A simple and straightforward system design approach is favoured, thereby facilitating operator understanding of plant response and minimizing configuration changes. The four train configuration offers the possibility of extended periods of maintenance on parts or even entire systems, useful for preventive maintenance and repair work during normal operation.

The safety injection systems, for which an overview is presented in Table 4.5-I, feature an In-containment refuelling water storage tank (IRWST) located at the bottom of the containment and provide injection in both hot and cold legs of the RCS. During design basis accidents the low-head safety injection (LHSI) system transfers the decay heat to the ultimate heat sink via heat exchangers. The containment spray system foreseen is only used for heat removal in case of severe accidents. The primary side safety systems are designed in accordance with stringent acceptance criteria to ensure limited fuel damages, even in case of large breaks. The delivery head of the medium head safety injection (MHSI) system will be adjusted below the steam generator relief and safety valve set points.

In case of a steam generator tube rupture, the affected steam generator will be isolated on the secondary side. After the initial transient, the primary and secondary pressures will equalize at a level below the set points of the safety valves in this steam generator, limiting to negligible levels the radiological releases.

Table 4.5-I. Organization of safety injection systems

MHSI Medium Head Safety Injection system	4 trains, cold side injection
Accumulators	4 accumulators, cold side injection
LHSI Low Head Safety Injection system	4 trains, cold side injection (short term); combined hot /cold injection (long term)
RHR Residual Heat Removal system	2 trains, independent from safety injection systems
IRWST In-containment Refuelling Water Storage Tank	Storage of borated water inside containment

Table 4.5-II. Diverse system functions for major safety systems

Complete failure of	Diverse system function		
MHSI Medium Head Safety Injection system	Fast secondary side pressure relief +	Accumulator injection system +	LHSI Low Head Safety Injection system
LHSI Low Head Safety Injection system	MHSI Medium Head Safety Injection system +	Containment heat removal system	(for small breaks)
RHR Residual Heat Removal system	Secondary side heat removal system, or	LHSI Low Head Safety Injection system	
Fuel pool cooling system	Fuel pool heat-up (boiling) +	Coolant fill up	
Secondary side heat removal system	Primary side feed and bleed		

A separate residual heat removal (RHR) system is provided, arranged in two trains and installed inside the containment to minimize risk of containment bypass. Adequate additional redundancy and diversity in decay heat removal is ensured by two of the four LHSI trains, which can serve in an RHR mode at low RCS temperature.

The function of any one of the safety systems can be accomplished by another diverse system (or group of systems) in the event malfunctions, as shown in Table 4.5-II.

In the EPR design efforts have been devoted to prevent high pressure core melt scenarios. Prevention of such scenarios implies need of a highly reliable secondary side heat removal system.

Detailed investigations of active versus passive systems have led to selection of an active emergency feedwater system with diversified power supply to the pumps to achieve a very high reliability. This system consists of four separate and independent trains, each with an emergency feedwater pump supplying feedwater to one of the four steam generators.

Safety injection systems

The safety injection systems mitigate loss of coolant accidents of all sizes, specific non-LOCA events, such as main steam line breaks and sequences leading to feed and bleed. The systems ensure heat removal, coolant inventory and reactivity control.

The medium head safety injection (MHSI) system feeds into the cold legs of the reactor coolant system. The shut-off head of the system is 8.0 MPa. This shut-off head is sufficient to cope with all LOCA related requirements, since a reliable secondary side partial cooldown is provided via safety-grade main steam relief valves. In conjunction with two small letdown lines, connecting two different hot legs with the IRWST, the MHSI system can be used for safety-grade boration during design basis accidents (PCC-2-4).

In addition to the medium head injection system, hot leg accumulator injection is provided to cope with large and intermediate break sizes. Four accumulators with a volume of 50 m^3 each are provided, each directly assigned to one hot leg. The response pressure of the accumulators is designed to 4.5 MPa. The low pressure injection system with a shut-off head of 2.5 MPa together with the respective accumulator water volume provides a continuous water injection capability.

The low pressure injection system feeds initially into the cold leg. In order to stop the core outlet steaming and the steam release to the containment, a switching to combined injection into the hot and cold legs after 1-2 hours is foreseen. The injection pressure of 2.5 MPa offers advantages for feed and bleed operation and supports accumulator injection in an optimum way for a large spectrum of break sizes.

In addition to these accident mitigation functions, the low pressure injection system is designed to be a backup for the operational residual heat removal system at low RCS temperatures.

In-containment refuelling water storage tank

The In-containment refuelling water storage tank provides the source for emergency core cooling water and is located inside the containment between the reactor cavity and the missile protection cylinder on the bottom level of the containment. In the case of loss of coolant accidents, or in feed and bleed situations, the safety injection system draws from the In-containment refuelling water storage tank. The water steam mixture escaping through the leak, and through the bleed valve, respectively, is returned to the tank. In the case of severe accidents the In-Containment water storage tank will provide the cooling water for flooding the spread molten corium.

In addition, the storage tank provides water for the operational function of flooding the reactor pit and the pools during refuelling.

Emergency feedwater system

The emergency feedwater system which is shown on figure 4.5-6, consists of four separate and independent trains, each providing injection to one of the four steam generators. Each emergency feedwater pump takes suction from an emergency feedwater tank. These tanks and the systems are located in the four divisions of the safeguard buildings.

The EFW system does not have any operational functions. The four emergency feedwater pumps will be driven by electric motors which are emergency power supplied; in addition, two of them are connected to small diversified diesels so that the probability of common cause failure of all emergency power supplies is reduced to the maximum extent.

For start-up and shutdown a dedicated system is installed. This system is automatically started in case of loss of main feedwater and provides an efficient feature to minimize the need for the EFW system.

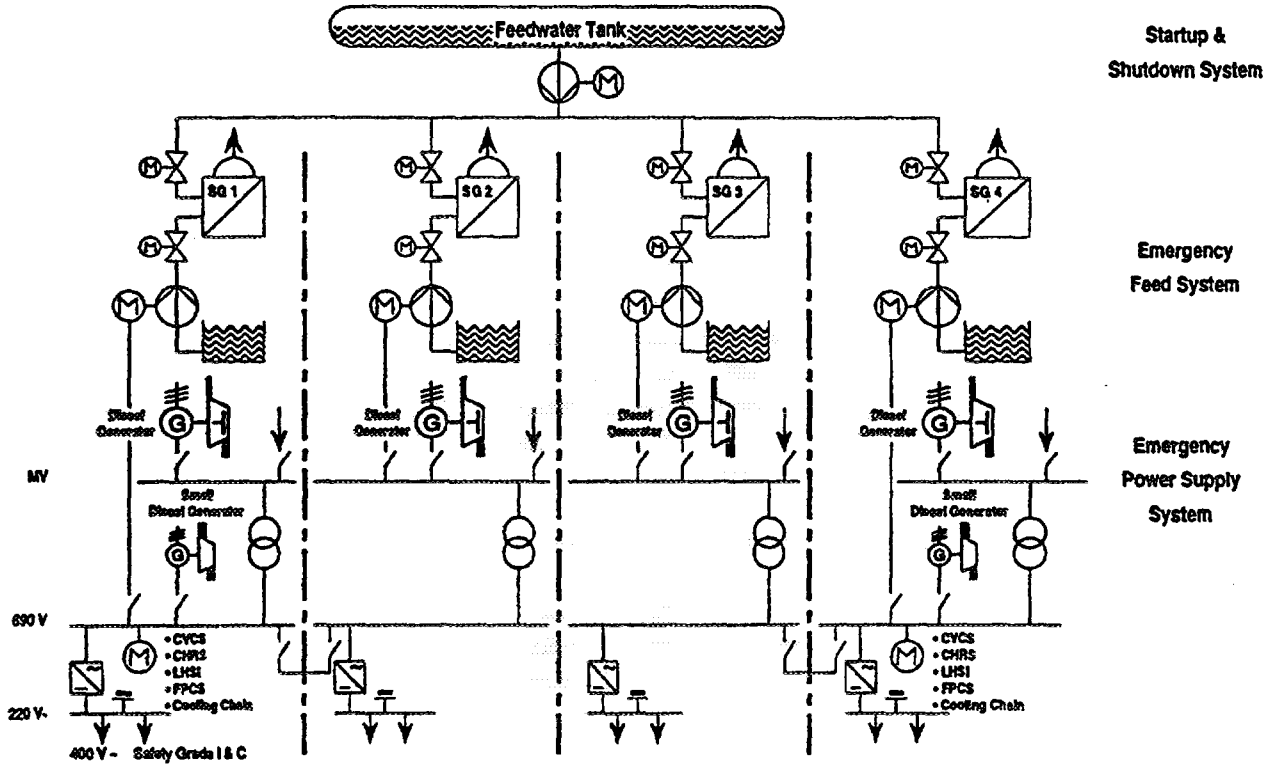


Figure 4.5-6 EPR - Configuration of emergency feedwater system

The emergency feedwater system transfers the residual and latent heat from the reactor coolant system via the steam generators to the atmosphere as long as the steam generator saturation temperature is above 150 °C, following any plant incident or accident other than those reactor coolant boundary ruptures for which complete residual heat removal by the safety injection system is possible (i.e. moderate to large size loss of coolant accidents). Following a LOCA in the size range which implies that not all core residual heat is released through the break flow and that at least a portion of the heat must be removed via the steam generators, the emergency feedwater system ensures sufficient water supplies to the steam generators.

In the case of a steam generator tube rupture, the emergency feedwater system removes the heat via the intact steam generators. The pressure in the affected steam generator is allowed to increase so as to reduce and eventually eliminate the break flow from primary to secondary side; the maximum pressure will remain at a level below the opening setpoint of the steam generator relief and safety valve.

The emergency feedwater system keeps the water inventory of at least one steam generator above an adequate level to maintain primary to secondary heat transfer, assuming a single failure.

Safety-grade, normally locked closed headers on pump discharge and on tank side ensure feed of all SGs and use of all water masses stored even in case of a single failure. These headers can only be opened at the system after sufficient grace period.

After a small break LOCA or a steam generator tube rupture, the emergency feedwater system provides enough cooldown capability so that the saturation pressure of the steam generator of 6.0 MPa is reached within a time span consistent with safety injection system performance requirements, and the radioactivity release limits for steam generator tube rupture, assuming a single failure.

The emergency feedwater system shall provide sufficient heat removal capacity and autonomy to ensure continued removal of decay heat for 24 hours with a final reactor coolant system temperature not exceeding nominal hot shutdown conditions. This shall be accomplished also under the assumption that no electric power is supplied from external sources and that the ultimate heat sink is not available.

Residual heat removal system

The residual heat removal system is designed to transfer residual heat from the reactor coolant system via the cooling chain consisting of the component cooling water system and service water system to the ultimate heat sink, when heat removal via the steam generators is not sufficient. Furthermore, it ensures continued heat transfer from the reactor coolant system or from the In-containment refuelling water storage tank during cold shutdown or refuelling conditions.

The residual heat removal system combines operational and safety functions. The safety function implies that the residual heat removal system, in conjunction with the component cooling water system and the service water system, shall maintain the reactor coolant system core outlet and hot leg temperature below 180°C following a reactor shutdown, assuming a single failure and the maximum design temperature of the service water. The operational function requires the residual heat removal system to be capable of cooling down the reactor coolant system to 50°C following a reactor shutdown, with at least one reactor coolant pump in operation and considering the maximum design temperature of the service water.

The residual heat removal system consists of two separate and independent trains located inside the containment. Each of the two pumps draws water from a separate line connected to a hot leg of the reactor coolant system. The pump discharge is routed via heat exchangers to a cold leg of the reactor coolant system. A bypass line of the heat exchanger is provided to allow control of the cool-down rate. The residual heat removal system heat exchangers are cooled by the component cooling water system train, which is located in the same division as the associated residual heat removal train. Switch-over from secondary side cooling to residual heat removal cooling is foreseen at an average reactor coolant system temperature between 150 and 180°C. During normal operation, only the inside RHR trains are used for cooldown and cold shutdown. Back-up trains are not used unless the RCS temperature is below 100°C.

In case of a total loss of both residual heat removal trains, the residual heat removal function can be taken over by two of the four low head safety injection trains, which function as a back-up for the residual heat removal system. Separate suction line connections to the reactor coolant system are foreseen for this purpose so that a total of four suction connections is available. The two low head safety injection trains that are designed to be back-up to the residual heat removal system, are assigned to different divisions than the corresponding residual heat removal system trains.

4.5.6.3 Severe accidents (beyond design basis accidents)

Severe accident mitigation strategy

The EPR design enhances accident prevention and reduces the residual risk for the public and the environment by reducing possible releases of radioactive material and radiological consequences arising from severe core damage sequences

The licensing authorities tend to take risk reductions in the event of more and more accidents into account, in particular with respect to possible need for relocation or evacuation of the population beyond the immediate vicinity of the plant, and restrictions on the use of foodstuff beyond the first year harvest.

The design target of the EPR is that off-site emergency response actions (population evacuation or relocation) shall be restricted to the nearby plant vicinity. To this end, maintaining the integrity of the containment is utterly important, and this will be achieved by:

- Avoidance of early containment failure or bypass.
- Cooling of the corium in the containment and retention of fission products by water covering.
- Preservation of containment functions, such as low leak rates, reliable containment isolation function and prevention of basemat melt-through, ultimate pressure resistance to cope with energetic events.
- Pressure reduction inside the containment by dedicated heat removal
- Collection of unavoidable containment leakages in the annulus atmosphere and release via the stack after filtration.

Severe accident prevention and mitigation features

The EPR strategy includes both preventive measures and mitigating features:

- prevention of high pressure core melt situations, by ensuring a high reliability of the decay heat removal systems, complemented by pressurizer relief valves. The depressurization eliminates high pressure failure of the RPV and the danger of direct containment heating with the potential consequence of early containment failure. The consequences of an instantaneous full cross-section break of the RPV at a pressure of about 2.0 MPa are nevertheless taken into account for the layout and support design.
- prevention repectively reduction of the hydrogen-concentration in the containment by catalytic H_2 -recombiners and, if necessary, by selectively arranged igniters. The prevention of molten core-concrete interaction contributes to reducing the amount of hydrogen.
- prevention of ex-vessel steam explosions endangering the containment integrity by minimizing the amount of water in the area where the corium is spread.
- prevention of a molten core-concrete interaction by spreading the corium in a spreading compartment provided with a protective layer (Figure 4.5-7), and
- connection of this spreading compartment to the reactor pit via a melt discharge channel which slopes towards the spreading compartment. This channel is closed by a steel plate (possibly covered with refractory material) which will resist melt-through for a certain time, in order to accumulate the melt in the pit.
- provisions for connecting the spreading compartment with the In-containment refuelling water storage tank (IRWST) for water flooding after spreading; these pipe connections are closed during normal operation.

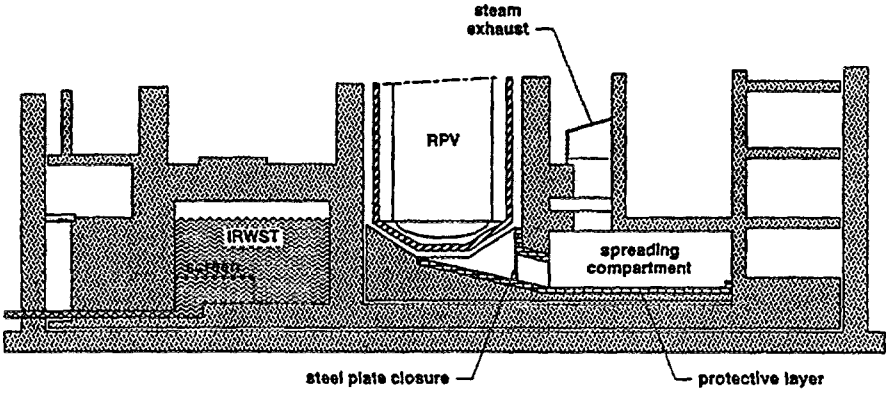


Figure 4.5-7 EPR - Retention of molten corium

- dedicated basemat cooling system supplied by the containment heat removal system to prevent high temperature loadings in the basemat.

4.5.7 Plant layout

The plant layout is governed by a number of principles derived from the huge experience gained through the construction and operation of the French and German nuclear power programmes with an installed capacity of more than 100 000 MW. The proven design contributes significantly to the the economic viability of the nuclear power plant.

4.5.7.1 Buildings and structures, including plot plan

The general layout of the EPR plant is shown on Figure 4.5-8. The reactor building with the containment is surrounded by the safeguard and fuel buildings which contain the safety systems. Most of the safety-grade systems are designed with a four-fold redundancy, arranged in four independent divisions with complete physical separation. Each division comprises a low head injection system with the related intermediate cooling system, a medium head injection system and an emergency feedwater system.

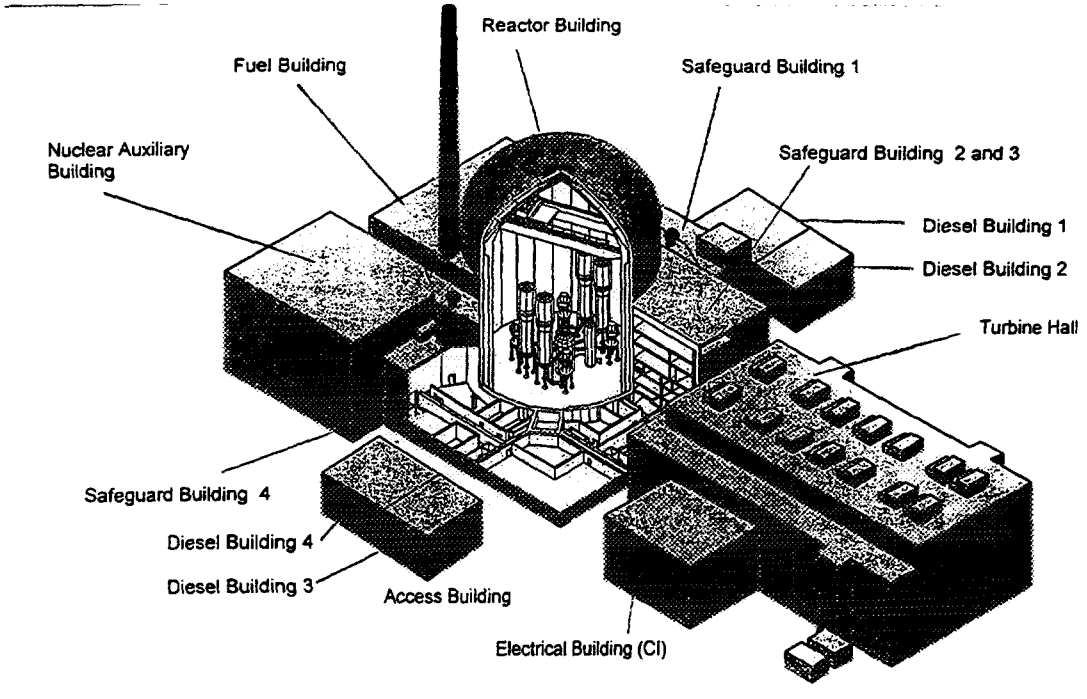


Figure 4.5-8 EPR - Plot plan

Curve C1 is to be used for the design of:

- components against induced vibrations. Impact is assumed to be in the centre of respective outer protection wall;
- outer protection walls against penetration without "scabbing".

Curve C2 is to be used for the design of:

- outer protection walls in accordance with Eurocode 2 "Ultimate Limit State". "Scabbing" is permitted as far as safe shutdown of reactor is ensured and core melt is prevented.

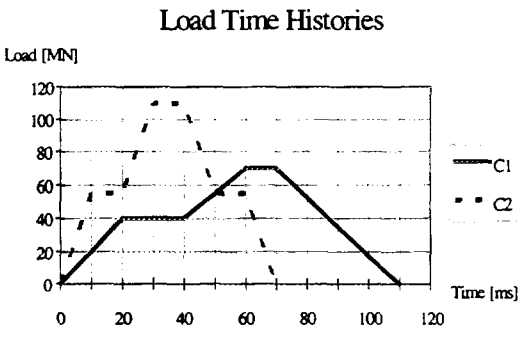


Figure 4.5-9 Design loading of aircraft crash

The related electrical systems as well as the instrumentation and control systems are also allocated to these divisions but on a higher building level.

Other plant buildings, such as the access building and the nuclear auxiliary building, are located in close contact with the safeguard and fuel buildings, whereas the turbine building and the associated conventional electrical building are built separated from the reactor building complex and arranged so that the reactor building is located in the projection of the turbine generator shaft.

Design requirements

The plant is designed to withstand the impacts of internal and external events, as specified below. With respect to earthquake and explosion pressure waves, the buildings and structures have been strengthened so that the function of safety-grade equipment will not be jeopardized by collapsing structures, and that the equipment itself must withstand the dynamic effects inside the buildings. For protection against airplane crash, safety-related equipment is located in bunkers, or redundant portions will be geographically separated from each other so that only one train need be considered as impacted.

Aircraft crash: The assumptions with respect to aircraft crash are based on a probabilistic risk assessment, since statistical data are sufficiently representative and possible events are quite well known. The design load case for an aircraft crash as it has been defined by GRS/RSK on 10 January 1995 is described in Figure 4.5-9. Earthquake: Provisionally, the seismic design is based on the spectrum defined in the EUR (European Utility Requirements), scaled to 0.25g, for the free field level of horizontal movement, for a wide range of soil conditions.

Explosion pressure wave: A review of probability data regarding the risk of an impact by an external explosion for various sites indicates that they are closely related to the industrial environment of each site.

For the EPR, the design is based on an incoming pressure wave with a maximum over-pressure of 10 kPa (100 mbar) as depicted in Figure 4.5-10. The level of protection will be verified on a site by site basis.

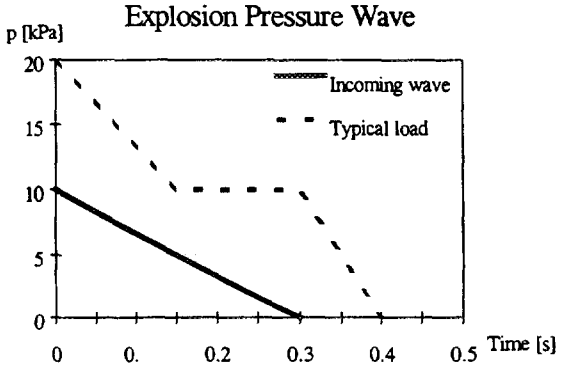


Figure 4.5-10 Explosion pressure wave

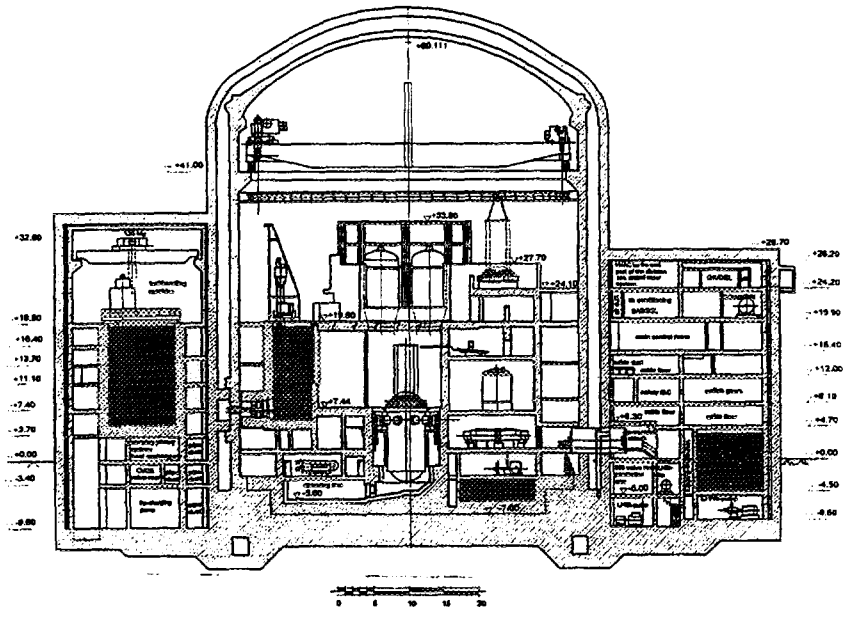


Figure 4.5-11 EPR - Building arrangement, section A-A

Internal hazards: The loads from internal events (e.g. fire loads, missile loads, jet impingement loads, flooding effects) are design-dependent and have not been fixed yet. For the overall plant layout they should be minimized and easy protection measures should protect sensitive equipment.

The protection against external and internal hazards includes the divisional separation of safety-grade systems and the physical protection of the containment enclosing the reactor coolant pressure boundary. By these means, the risk of inadmissible releases or common-mode failures of safety-grade system will be consistent with the deterministic design basis and the probabilistic targets of the EPR.

Besides the requirements concerning severe accident mitigation, the application of radiation protection principles influences the plant layout significantly. The EPR is designed for a separation of hot (controlled) and cold (not controlled) areas.

The reactor building and the fuel building are classified as hot zones. Within the safeguard buildings, the safety injection system part is arranged in the inner areas, which are classified as hot zones, whereas the component cooling and emergency feedwater systems are installed in the outer areas which are classified as cold zones.

4.5.7.2 Reactor building

The reactor building (Figures 4.5-11 and -12, -13) is the central building of the plant complex. In essence, it coincides completely with the containment, and thus, the following description of the containment covers also the reactor building.

4.5.7.3 Containment

Adoption of a double concrete containment design was decided for the EPR. The particular design concept uses, for the inner containment wall, the prestressed concrete technology. The leak-tightness requirement of less than 1% volume per day can be ensured without provision of a containment liner. The outer wall, in reinforced concrete, completes the double containment arrangement.

To ensure containment leak-tightness, systems for isolation and retention and control of leakages are required. Leakages through the inner containment wall are released via the annulus air extraction system. Openings for personnel access or equipment supply to the inside of the containment are permanently closed hatches or air locks with double sealings on both sides.

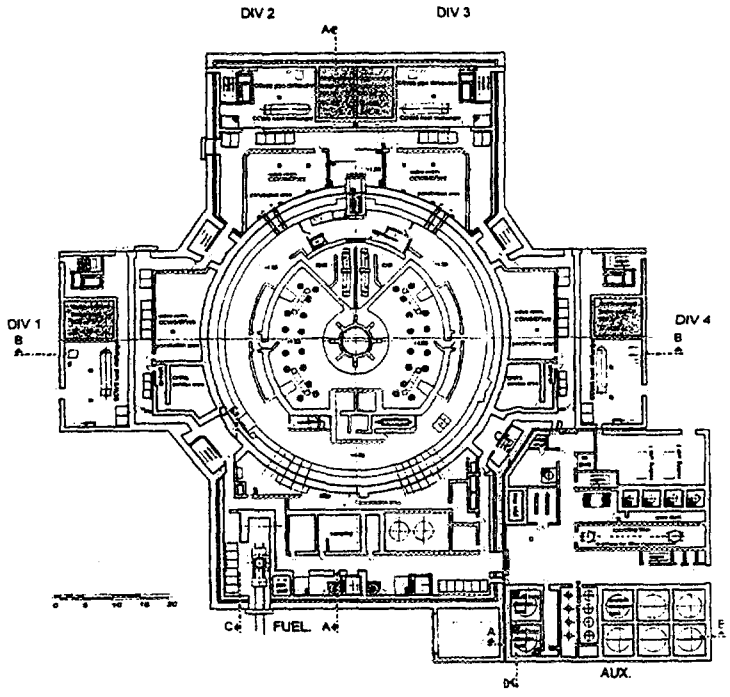


Figure 4.5-12 EPR - Building arrangement, plan view ± 0,00 m

This concept is also applied for penetrations of the HVAC systems. Fluid systems penetrating the containment are provided with double isolation valves, inside and outside the containment.

The structural integrity of the containment is protected by the thermal inertia of the concrete structures inside the containment (absorbing heat), and the safety injection system and the containment heat removal system (removing heat).

Accumulation of combustible gases, especially hydrogen, is controlled. Furthermore, the basemat in the spreading compartment is protected by protective layers and the dedicated cooling system fed by the containment heat removal system against elevated temperatures resulting of a core melt.

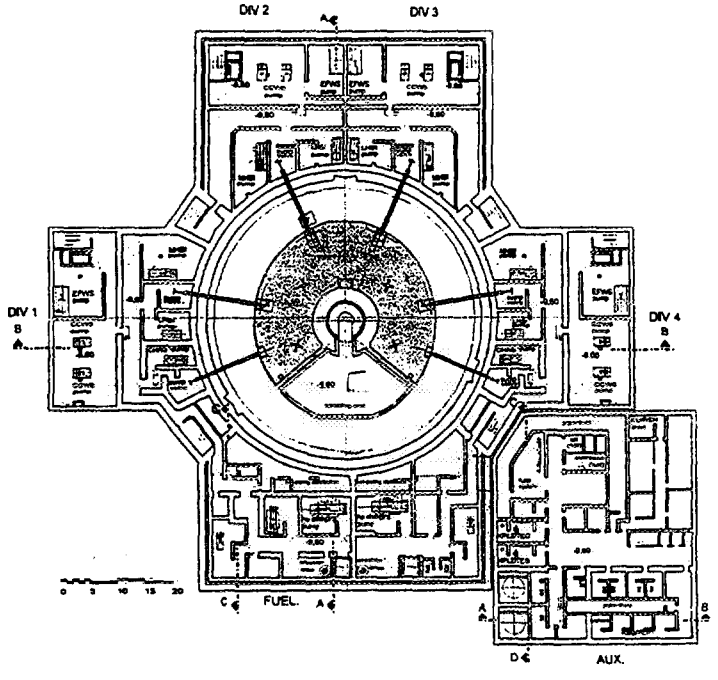


Figure 4.5-13 EPR - Building arrangement, plan view - 9,60 m

The severe accident conditions described in the previous sections, lead to more severe design conditions compared to the existing plants, and will thus result in an extrapolation of the design parameters, In this respect, the most important factor is the increased design pressure, which was defined as 650 kPa (6.5 bar abs).

The prestressed concrete inner wall will also ensure capability to perform an integral leakage pressure test in air at design pressure, thus providing positive proof of containment structural and leak-tightness capability for the entire range of pressure of all severe accident scenarios.

Containment integrity

The requirement on limitation of the radiological consequences to the environment of the plant even under severe accident conditions implies strong demands on the containment as the last barrier for radioactive releases.

The maximum pressure and temperature reached during the most severe transient shall not exceed the design values, e.g., the design pressure of 650 kPa. The leak rate of the inner containment shall be lower than 1% of the containment volume per day under accident pressure.

It is important to control the formation of hydrogen, e.g., by zirconium-water reaction, and deflagration or detonation, the formation of non-condensable gases, e.g., as a consequence of corium-concrete interaction, and the residual heat generation by the molten core.

For residual heat removal from inside the containment after severe accidents the containment heat removal system is provided. Its prime function is to limit the pressure increase inside the containment below the design pressure, and to decrease this pressure to restrict the fission product releases through postulated containment leaks. For this purpose, a spray system with heat exchangers has been selected - with all active components located in a special compartment outside the containment.

The containment integrity and the core melt generated following severe accidents need to be controlled. Therefore the reactor pit design has been modified compared to earlier designs. The reactor pit bottom is by a slope connected to a spreading area which is provided to collect the core debris and separate it from the in-containment water storage tank to avoid steam explosion. In a later stage of the accident, water ingress is provided by dedicated melting plugs that allow water to cool the molten core material by passive means. The generated steam is condensed by a containment cooling system exclusively provided for these accident sequences.

Interaction between concrete and the molten core material is prevented by a high temperature resistant protective layer on the reactor pit floor and the spreading area.

4.5.7.4 Turbine building

As shown in Figure 4.5-8, the turbine building is directly adjacent to the reactor building. In the French version, the main characteristics of the building are:

- the building is a rather conventional metallic beam building, with a concrete basemat;
- the only other concrete structure is the turbine generator support;
- no safety-related equipment is located in the building;
- the turbine generator shaft is projecting from the reactor building to limit missiles risks;
- the high-medium pressure part of the turbine is located near the reactor building, while the low-pressure stages are at the switchyard side of the building;
- the condenser, the condensate pumps and the startup pumps are located at the very bottom of the building;
- the electric motor-driven main feedwater pumps and high pressure feed heaters are located at ground level;
- the turbine, generator, and dryer-reheater units are located at the top of the buliding, together with the feedwater tank;

- the generator is connected to the main (step-up) transformer located outside the building; and
- auxiliary transformers, provided for auxiliary power supply in case of incidents on the grid, are connected between the main transformer and the line breaker (high voltage supply).

4.5.7.5 Other buildings

Safeguard and fuel buildings

Connections between the safety systems and the reactor coolant system are made as short as possible. The individual trains of the safety systems are arranged radially to the primary loops. Each train of the safety systems is protected against internal hazards from one train to another by location in specific separated areas, called divisions. The four trains of the safety systems are located in four safeguard buildings that surround the reactor building. The spent fuel pool is located in the fuel building which is separated from the safety divisions.

The reactor building, the fuel building and the four safeguard buildings are protected against external hazards, such as earthquake and explosion pressure wave. All these buildings are situated on a common raft.

Protection against aircraft crash is achieved by bunkerization of the safeguard buildings 2 and 3, the reactor building and the fuel building. The main control room and the remote shutdown station are also located in these bunkered safeguard buildings. The inner building structures of these buildings are decoupled from the outer shell in order to minimize induced vibrations.

The safeguard buildings 1 and 4 are not bunkered but geographically separated, so that only one division can be affected, the other remaining operable.

The storage pool for spent fuel assemblies is located outside the containment, facilitating fuel transport cask loading and unloading for the transport of fresh fuel to the plant and of spent fuel away from the plant. A transfer tube in the containment wall connects the inside of the containment with the fuel building.

The primary system is arranged symmetrically. Concrete walls are provided between the loops and between the hot and cold legs of each loop to provide protection against consequential failures. The pressurizer is located in a separate compartment. A concrete wall around the entire primary system protects the containment from missiles and reduces the radiation from the primary system to the surroundings.

A water pool for storage of the upper core internals during refuelling, and for the entire core internals during inspection, is provided inside the containment for radiation protection reasons.

Nuclear auxiliary building

This building houses mainly

- Boron recycle system (coolant and demineralized water storage, coolant treatment and coolant purification)
- Fuel pool purification system
- Gaseous waste system
- Steam generator blowdown system (including purification)

Diesel buildings

The four diesel buildings constitute the four redundancies of the emergency power supply system. Diesel buildings 1/2 and 3/4 are located at opposite positions adjacent to the reactor building.

The diesel buildings 1 and 4 are housing also the smaller diesel generator sets.

4.5.8 Technical data

General plant data		
Power plant output, gross	1 500	MWe
Power plant output, net		MWe
Reactor thermal output	4 250	MWt
Power plant efficiency, net		%
Cooling water temperature		°C
Nuclear steam supply system		
Number of coolant loops	4	
Primary circuit volume, including pressuriser	380+75	m^3
Steam flow rate at nominal conditions		kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	288/72.5	°C/MPa
Feedwater temperature/pressure		°C/MPa
Reactor coolant system		
Primary coolant flow rate	21 900	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	291	°C
Coolant outlet temperature, at RPV outlet	327	°C
Mean temperature rise across core		°C
Reactor core		
Active core height	4.2	m
Equivalent core diameter		m
Heat transfer surface in the core		m^2
Fuel inventory		t U
Average linear heat rate	15.5	kW/m
Average fuel power density		kW/kg U
Average core power density (volumetric)		kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		

Fuel material	Sintered UO_2	
Fuel assembly total length	4 800	mm
Rod array	square, 17x17	
Number of fuel assemblies	241	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	25	
Number of spacers	9	
Enrichment (range) of first core	≤ 5.0	Wt%
Enrichment of reload fuel at equilibrium core		Wt%
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel	60 000	MWd/t
Cladding tube material	Zr 4	
Cladding tube wall thickness	0.725	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly		kg
Active length of fuel rods		mm
Burnable absorber, strategy/material		
Number of control rods	81	
Absorber rods per control assembly		
Absorber material	Ag-80, In-15, Cd-5	
Drive mechanism	Magnetic jack	
Positioning rate [10x75 or 750 mm/min]	75	steps/min
Soluble neutron absorber		
Reactor pressure vessel		
Cylindrical shell inner diameter	5 000	mm
Wall thickness of cylindrical shell	250 + 6	mm
Total height		mm
Base material: cylindrical shell	16MND5/20MnMoNi55	
RPV head		
liner	stainless steel	
Design pressure/temperature	17.6/	MPa/°C

Transport weight (lower part)		t
RPV head		t

Steam generators

Type	U-tube heat exchanger	
Number	4	
Heat transfer surface	7 300	m^2
Number of heat exchanger tubes		
Tube dimensions		mm
Maximum outer diameter		mm
Total height		mm
Transport weight		t
Shell and tube sheet material	16MND5/20MnMoNi55	
Tube material	Incoloy 800 or Inconel 690	

Reactor coolant pump

Type	Single-stage, centrifugal pump	
Number	4	
Design pressure/temperature		MPa/°C
Design flow rate (at operating conditions)	5 475	kg/s
Pump head		
Power demand at coupling, cold/hot		kW
Pump casing material	Stainless or Ferritic with cladding 3 seals	
Pump speed		rpm

Pressuriser

Total volume	~ 75	m^3
Steam volume: full power/zero power		m^3
Design pressure/temperature	17.6/	MPa/°C
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter		mm
Total height		mm
Material	16MND5/20MnMoNi55	
Transport weight		t

Pressuriser relief tank (if any)

Total volume	m^3
Design pressure/temperature	MPa/°C
Inner diameter (vessel)	mm
Total height	mm
Material	
Transport weight	t

Primary containment

Type	prestressed concrete	
Overall form (cyl.)	cylindrical	
Dimensions (diameter/height)	48/	m
Free volume	90 000	m^3
Design pressure/temperature (DBEs)	~ 650/	kPa/°C
(severe accident situations)	~ 650/	kPa/°C
Design leakage rate	< 1	vol%/day
Is secondary containment provided?	yes, reinforced concrete (APC-Protection)	

Reactor auxiliary systems

Reactor water cleanup,	capacity	kg/s
	filter type	
Residual heat removal,	at high pressure	kg/s
	at low pressure	kg/s
Coolant injection,	at high pressure	kg/s
	at low pressure	kg/s

Power supply systems

Main transformer,	rated voltage	kV
	rated capacity	MVA
Plant transformers,	rated voltage	kV
	rated capacity	MVA
Start-up transformer	rated voltage	kV
	rated capacity	MVA
Medium voltage busbars (6 kV or 10 kV)		
Number of low voltage busbar systems		
Standby diesel generating units: number		

rated power		MW
Number of diesel-backed busbar systems		
Voltage level of these		V ac
Number of DC distributions		
Voltage level of these		V dc
Number of battery-backed busbar systems		
Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)		
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HMP/ 3 LP	
Turbine speed	1500	rpm
Overall length of turbine unit	68.77	m
Overall width of turbine unit	18	m
HP inlet pressure/temperature		MPa/°C

Generator

Type	two-pole, synchronous	
Rated power	1900	MVA
Active power	1600	MW
Voltage	20	kV
Frequency	50	Hz
Total generator mass	> 500	t
Overall length of generator		m

Condenser

Type [for once-through operation]	modular	
Number of tubes	84 400	
Heat transfer area	89 763	m^2
Cooling water flow rate	43	m^3/s
Cooling water temperature	13	°C
Condenser pressure	51.5	hPa

Condensate pumps

Number	2x50%	
Flow rate	670	kg/s
Pump head	365	m WG
Temperature	40	°C
Pump speed	1500	rpm

Condensate clean-up system

Full flow/part flow		
Filter type		

Feedwater tank

Volume	830	m^3
Pressure/temperature	10/180	MPa/ °C

Feedwater pumps

Number	3x50%	
Flow rate	2x1200	kg/s
Pump head	850	m WG
Feedwater temperature	230	°C
Pump speed [variable, full speed =]	4700	rpm

Condensate and feedwater heaters

Number of heating stages	6 (4 LP + 2 HP)
Redundancies	2

4.5.9 Project status and planned schedule

The joint programme agreed upon between France and Germany has already been running some years. In the recent years, the work was mainly concentrated on the conceptual design and the overall safety strategy. Since 1993, the concept proposals are subjected to assessment of the authorities of both countries. The authorities have issued a report "GPR - RSK Proposal for a Common Safety Approval for Future PWRs"; this report forms the background for the joint assessment of the safety features of the EPR.

Two major milestones have been achieved:

1) The safety authorities of both countries have positively accomplished this assessment of the main safety issues.
2) EdF and the German utilities have, on 23 February 1995, awarded a contract to NPI and its parent companies concerning the Basic Design Engineering. This work is scheduled to last 2½ years. Within this time period, all documentation needed for an application for a construction license will be prepared. In this context, the measures to be foreseen to cope with severe accidents represent an important item. The integrity of the containment has to be ensured, and direct leakage from the containment into the the environment must be prevented.

The basic design engineering phase will be followed by detail design activities. At the same time, the issuance of a general license is expected. At the end of the century, NPI will be ready for marketing of the EPR and construction of the first plant may start in the beginning of year 2000. No specific site has yet been selected.

References

A comprehensive presentation of the EPR plant design and project took place at the Palais de la Musique et de Congrés, Strasbourg, 13 - 14 November 1995, organized jointly by Société Francaise d'Energie Nucleaire, France (SFEN) and Kerntechnische Gesellschaft, Germany (KTG).

The lectures given are compiled in the proceedings entitled "The EPR Project", Strasbourg, Palais de la Musique et de Congrés, 13 - 14 November 1995.

4.6 ESBWR, GE, USA

4.6.1 Introduction

The design of the ESBWR (European Simplified Boiling Water Reactor) of General Electric represents a design for a 1190 MWe power plant, based on the earlier work done on the 670 MWe SBWR power plant. The rated thermal output of the reactor core is 3613 MWth. The ESBWR and the SBWR incorporate innovative, yet proven, concepts to further simplify an inherently simple direct cycle nuclear plant. The features selected are all proven concepts in large nuclear power plants.

The ESBWR design objectives include: 60 year plant life from full power operating license date, 89% or greater plant availability, 24 month refuelling interval, personnel radiation exposure limit of 100 manrem/year, providing safety related functions primarily through passive means, core damage frequency of less than 10^{-5} per reactor year, limiting significant release frequency, from all events (internal & external) to 10^{-6} per reactor year, and requiring no operator action on safety systems for 72 hours following a design basis accident to maintain the reactor and containment at safe stable conditions except for operator actions necessary to maintain main control room habitability.

The principal design criteria governing the ESBWR Standard Plant encompass two basic categories of requirements: they are related to either a power generation function or a safety related function.

General power generation design criteria

The plant is designed to produce electricity from a turbine generator unit using steam generated in the reactor. Heat removal systems are provided with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and abnormal operational transients. Backup heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage.

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

Control equipment is provided to allow the reactor to respond automatically to load changes and abnormal operational transients. Reactor power level is manually controllable. Interlocks or other automatic equipment are provided as backup to procedural control to avoid conditions requiring the functioning of safety related systems or engineered safety features.

General safety design criteria

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

The reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient. The reactor is designed with abundant core coolant flow so that there is high flow margin to prevent divergent oscillation of any operating characteristics considering the interaction of the reactor with other appropriate plant systems.

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

structures includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the plant site.

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

A containment is provided that completely encloses the reactor systems, drywell, and suppression chamber. The containment employs the pressure suppression concept. A reactor building is provided that encloses the containment. The containment and reactor building, in conjunction with other safety related features, limit radiological effects of design basis accidents to less than the prescribed acceptable limits. A perspective of the reactor building arrangement is shown in Fig. 4.6-1.

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

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

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

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

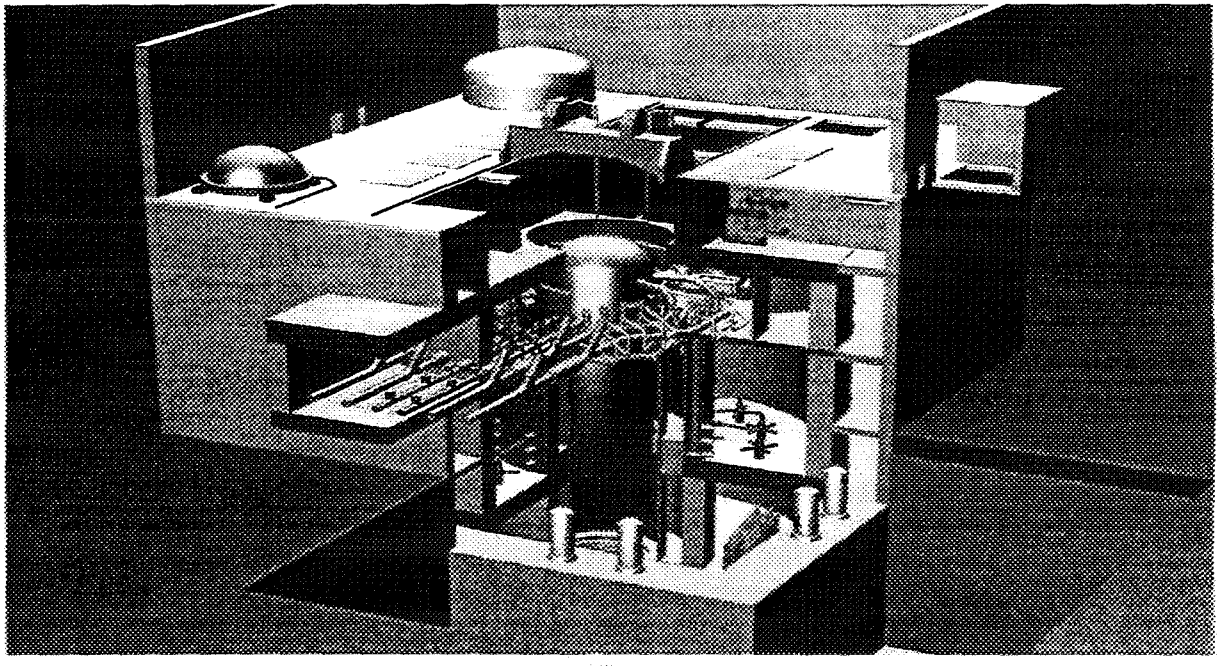


Figure 4.6-1 ESBWR - Reactor building arrangement

4.6.2 Description of the nuclear systems

4.6.2.1 Primary circuit and its main characteristics

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

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

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

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

The nuclear pressure relief system consists of safety/relief valves (SRVs) located on the main steam lines (MSLs) between the RPV and the inboard main steam isolation valve (MSIV). There are four SRVs per MSL. The SRVs provide overpressure and depressurization capability. Each of these protective functions are discussed below.

The automatic depressurization subsystem (ADS) consists of the 16 SRVs and 8 depressurization valves (DPVs) and their associated instrumentation and controls. The ADS quickly depressurizes the RPV in sufficient time for the gravity driven cooling system (GDCS) to inject flow and replenish core coolant to maintain core temperature below design limits in the event of a LOCA. The ADS maintains the reactor depressurized following a LOCA such that continued operation of GDCS is assured. Once actuated, the ADS can maintain the reactor in a depressurized state without the need for AC or DC power.

The SRVs and DPVs are actuated in groups of valves at staggered times as the reactor undergoes a relatively slow depressurization. This minimizes reactor level swell during the depressurization, thereby maximizing the residual coolant inventory in the RPV. The use of a combination of SRVs and DPVs to accomplish the ADS function provides an improvement in ADS reliability against hypothetical commonmode failures of otherwise nondiverse ADS components.

4.6.2.2 Reactor core and fuel design

The ESBWR core configuration consists of 1132 bundles. The inlet orifice of the peripheral bundles is restricted in order to preferentially force flow through interior, high power bundles. The rated core power is 3613 MWt, which corresponds to a 48 kW/l power density. The relatively low power density results in improved fuel cycle costs and greater manoeuvrability. Since the ESBWR is a natural circulation reactor, reactivity control is maintained by movement of control rods and by burnable poisons in the fuel.

Control rod drive system

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

The FMCRDs provide electric motor driven positioning for normal insertion and withdrawal of the control rods and hydraulic powered rapid control rod insertion (scram) in response to manual or automatic signals from the reactor protection system (RPS). Simultaneous with scram, the FMCRDs also provide electric motor driven run in of all control rods as a means of rod insertion that is diverse from the hydraulic powered scram. The hydraulic power required for scram is provided by high pressure water stored in the individual hydraulic control unit (HCU) assemblies. Each HCU is designed to scram up to two control rods. The HCUs also provide the flow path for purge water to the associated drives during normal operation. The CRDH subsystem supplies high pressure demineralized water which is regulated and distributed to provide charging of the HCU scram accumulators and purge water flow to the FMCRDs. The capacity of the pumps is sufficient to maintain RPV water level for small line break LOCAs. The CRD system is designed with the capability to provide makeup water to the RPV while at high pressure as long as AC power is available. If sensed reactor water level reaches Level 2 (approximately 2.5 m above the top of the active fuel), the CRD pumps run out in an effort to recover level.

There are 269 FMCRDs mounted in housings welded into the RPV bottom head. Each FMCRD has a movable hollow piston tube that is coupled at its upper end, inside the reactor vessel, to the bottom of a control rod. The piston is designed such that it can be moved up or down, both in fine increments and continuously over its entire range, by a ball nut and ball screw driven at a nominal speed of 30 mm/s by the electric stepper motor. In response to a scram signal, the piston rapidly inserts the control rod into the core hydraulically using stored energy in the HCU scram accumulator. The FMCRD design includes an electro mechanical brake on the motor drive shaft and a ball check valve at the point of connection with the scram inlet line. These features prevent control rod ejection in the event of a failure of the scram insert line. There would be a maximum of 135 HCUs, each of which provides sufficient volume of water stored at high pressure in a pre-charged accumulator to scram at least two FMCRDs at any reactor pressure.

4.6.2.3 Fuel handling and transfer systems

The fuel handling floor in the reactor building is serviced with a refuelling platform, and an auxiliary platform. Figure 4.6-2 shows the operating floor layout.

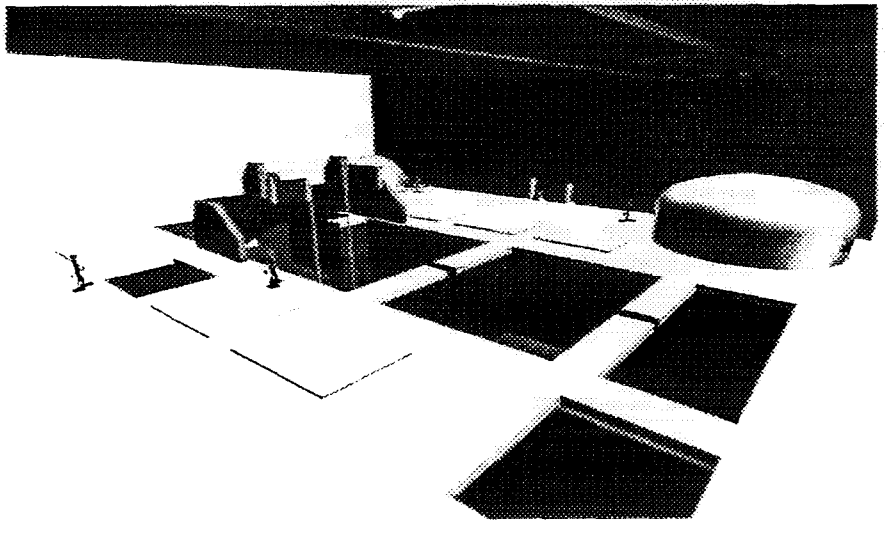


Figure 4.6-2 ESBWR Operating Floor

The refuelling platform is a gantry type crane which spans the reactor vessel cavity and fuel storage pool to handle fuel and perform other ancillary tasks. It is equipped with dual robotic arms for handling two fuel elements at a time. An auxiliary hoist is also provided. The platform is a rigid structure built to precise engineering standards to ensure accurate and repeatable positioning during the refuelling process. A programmed computer located above the refuelling floor controls the operational movements. Mechanical stops and interlocks provide the necessary operational limits.

The auxiliary platform is a low profile structure having its own track located on the fuel handling area floor. A removable section with mounted wheels is lowered to the reactor vessel flange level on which a special portable track is installed. Its primary purpose is to aid in open vessel servicing.

The spent fuel storage racks which have a storage capacity for fuel from five operating cycles plus one full core, consist of two tiered units constructed of stainless steel, laminated with neutron absorbing material. This ensures that a full array of loaded spent fuel will remain subcritical by 5%Δ/k, under all conditions.

4.6.2.4 Primary components

Reactor pressure vessel

The ESBWR reactor pressure vessel (RPV) assembly consists of the pressure vessel, removable head, and its appurtenances, supports and insulation, and the reactor internals enclosed by the vessel (excluding the core, incore nuclear instrumentation, neutron sources, control rods, and control rod drives). The RPV instrumentation to monitor the conditions within the RPV is designed to cover the full range of reactor power operation. The RPV, together with its internals, provides guidance and support for FMCRDs. Details of the RPV and internals are discussed below.

The RPV is a vertical, cylindrical pressure vessel consiting of rings welded together, with a removable top head, head flanges, seals and bolting. The vessel also includes penetrations, nozzles, shroud support, and venturi shaped flow restricters in the steam outlet nozzles.

The reactor vessel is 7.1 meters (23.3 ft.) inside diameter minimum, with a wall thickness of about 182 mm (7.2 in.) with cladding, and 25.4 m (83.3 ft.) tall from the inside of the bottom head (elevation zero) to the inside of the top head. The bottom of the active fuel location is 3750 mm (147.6 in.) from elevation zero and the active core is 2743 mm (108 in.) high. The overall RPV inside height of 25.4 m (83.3 feet) permits natural circulation driving forces to produce abundant core coolant flow. An increased internal flow path length relative to prior BWRs is provided by a long "chimney" in the space which extends from the top of the core to the entrance to the steam separator assembly. The chimney and steam separator assembly are supported by a shroud assembly which extends to the top of the core. The large RPV volume provides a large reservoir of water above the core, which insures remains covered following transients that result in feedwater flow interruptions or a LOCAs. This gives an extended period of time during which automatic systems or plant operators can reestablish reactor inventory control using any of several normal, non safety related systems capable of injecting water into the reactor. Timely initiation of these systems precludes the need for activation of emergency safety equipment. The large RPV volume also reduces the reactor pressurization rates that develop when the reactor is suddenly isolated from its normal heat sink. Reactor isolation leads to operation of the isolation condenser system (ICS) to control reactor pressure. The ICS eliminates the need to actuate safety related valves which result in the loss of RPV inventory to the suppression pool.

Reactor internals

The major reactor internal components include: (1) the core (fuel, channels, control rods and instrumentation); (2) the core support structures (shroud, shroud support, top guide, core plate, and integral control rod guide tube and orificed fuel support); (3) chimney and partitions; (4) chimney head and steam separator assembly; (5) steam dryer assembly; (6) feedwater spargers; (7) standby liquid control headers, spargers and piping assembly; and (8) incore guide tubes. Except for the Zircaloy in the reactor core, these

reactor internals are stress corrosion resistant stainless steels or other high alloy steels. The fuel assemblies (including fuel rods and channels), control rods, chimney head and steam separator assembly, steam dryers and incore instrumentation assemblies are removable when the reactor vessel is opened for refuelling or maintenance. In addition, the ESBWR is designed with removable internals.

The RPV shroud support is designed to support the shroud and the components connected to the shroud. The RPV shroud support is a series of horizontal brackets welded to the vessel wall which support the weight of the steam separator, chimney, core plate, top guide and the peripheral fuel bundles.

Reactor recirculation pumps

The ESBWR operates with natural circulation of the reactor coolant and no recirculation pumps are provided.

4.6.2.5 Reactor auxiliary systems

Plant service water system (PSWS) - The PSWS rejects heat from nonsafety related components in the reactor and turbine building to the environment. The PSWS consists of two independent and 100% redundant open loops continuously recirculating raw water through the heat exchangers of the reactor component cooling water system (RCCW) and the turbine component cooling water system (TCCW). Heat is rejected to the environment by mechanical draft cooling towers.

Reactor component cooling water system (RCCW) - The RCCW cools nonsafety related components in the reactor building and provides a barrier against potential radioactive contamination of the PSWS. The RCCW consists of two 100% capacity independent and redundant closed loops.

Make up water system (MWS) - The MWS is designed to supply demineralized water to the various nonsafety related systems that need demineralized water. The demineralization system design is dependent on site specific water quality.

Condensate storage and transfer system (CSTS) - The CSTS is a nonsafety related system that consist of two 100% pumps and lines taking suction from one storage tank that is the normal source of water for makeup to selected plant systems. The CSTS is also used for storage of excess condensate rejected from the condensate & feedwater systems and the condenser hotwell.

Chilled water system (CWS) - The CWS consists of two independent, nonsafety related, sub-systems: the reactor building CWS and the turbine building CWS. The CWS provides chilled water to the cooling coils of air conditioning units and other coolers in the reactor and turbine buildings. Each subsystem consists of two 100% capacity, redundant, and independent loops with cross-ties between the chilled water piping.

4.6.2.6 Operating characteristics

The ESBWR relies extensively on the lessons learned from operating BWRs regarding natural circulation operation, especially the GKN Dodewaard natural circulation reactor. The plant has been designed to maximize the core flow to ensure that there are huge margins to avoid unstable regions of operation. The core flow has been maximized by:

a) Using shorter fuel

b) Gaining high driving heads by using a tall chimney

c) Using improved low pressure loss steam separators.

The design features result in an average core flow per bundle that is almost 4 times higher than that for a forced circulation plant under similar conditions as shown in Figure 4.6-3. The use of natural circulation eliminates pumps, motors, controls, piping and many other components that could possibly fail and effect plant availability.

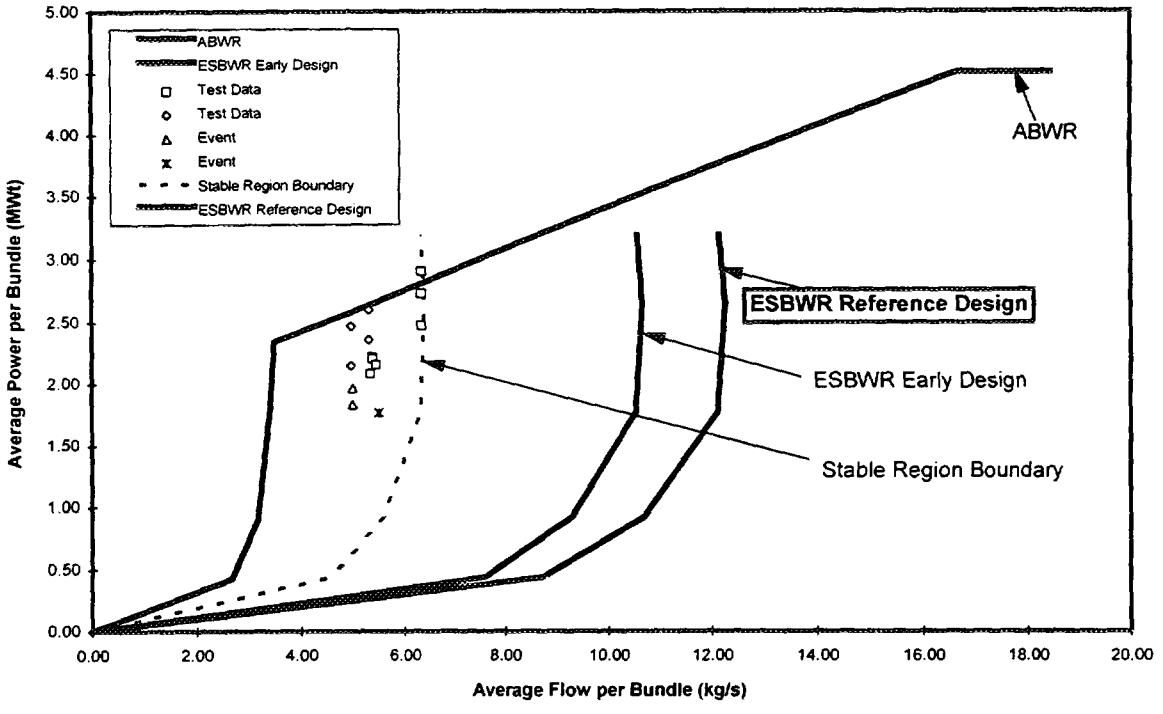


Figure 4.6-3 Comparison of average core flow per bundle for the ESBWR

Similar to the Dodewaard plant, the ESBWR is extremely simple to operate during startup and normal operation, and also has a very gentle transient response. This is because of the large reactor vessel and water inventory. A reactor isolation results in no loss of coolant inventory and no heating up of the containment or suppression pool. The transient and accident responses of the plant are discussed in some detail in Reference 9. The extensive experience of startup and normal operation of BWRs with features common to the ESBWR, including 25 years at Dodewaard, provide high confidence in the design.

4.6.3 Description of turbine generator plant system

4.6.3.1 Turbine generator plant

The main turbine

The main turbine is a tandem compound, four flow, 1320 mm (52 in) last stage buckets with one high pressure (HP) turbine and two low pressure (LP) turbines. The steam passes through an in-line high velocity moisture separator (HVS) and reheaters of advanced design prior to entering the LP turbines. Steam exhausted from the LP turbines is condensed and degassed in two series main condensers . The turbine uses steam at a pressure of 6.79 MPa (985 psia) from the reactor and rotates at 1500 RPM. Steam is extracted from several stages of each turbine and is used to heat the feedwater.

Turbine overspeed protection system

In addition to the normal speed control function provided by the turbine control system, a separate turbine overspeed protection system is included to minimize the possibility of turbine failure and high energy missile damage.

Turbine gland seal system

The turbine gland seal system (TGSS) provides steam sealing to the labyrinth seals of the high pressure and low pressure turbines and to the stem seals of the turbine stop valves, control valves and bypass valves. The system prevents the escape of radioactive steam from the turbine shaft/casing

penetrations and valve stems and prevents air in-leakage through subatmospheric turbine glands into the main condenser. The TGSS consists of a sealing steam pressure regulator, sealing steam header, a gland steam condenser, two full capacity exhaust blowers and associated piping, valves and instrumentation.

Turbine bypass system

A turbine bypass system (TBS) is provided which passes steam directly to the main condenser under the control of the pressure regulator. The TBS has the capability to shed 100% of the turbine generator rated load without reactor trip or operation of SRVs. The TBS does not serve or support any safety related function and has no safety design.

Main condenser

The main condenser is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system.

The main condenser, which does not serve or support any safety function and has no safety design basis, is a series arrangement two shells/four waterboxes type deaerating unit with each shell located directly beneath the respective low pressure turbines. Each shell has tube bundles through which circulating water flows. The condensing steam is collected in the condenser hotwells (the lower shell portion) which provide suction to the condensate pumps.

Since the main condenser operates at a vacuum, any leakage is into the shell side of the main condenser. Tubeside or circulating water in leakage is detected by measuring the conductivity of sample water extracted beneath the tube bundles. In all operational modes, the condenser is at vacuum and consequently no radioactive releases can occur.

Non-condensable gases are removed from the power cycle by the main condenser evacuation system (MCES). The MCES removes power cycle non-condensable gases, including the hydrogen and oxygen produced by radiolysis of water in the reactor, and exhausts them to the offgas system (OGS) during plant power operation, or to the turbine building ventilation system exhaust during early plant startup. The MCES establishes and maintains a vacuum in the condenser by the use of steam jet air ejectors during power operation, and by a mechanical vacuum pump during early startup.

4.6.3.2 Condensate and feedwater systems

The condensate and feedwater system (C&FS) consists of the piping, valves, pumps, heat exchangers, controls and instrumentation and the associated equipment and subsystems which supply the reactor with heated feedwater in a closed steam cycle utilizing regenerative feedwater heating. The four 1.1 MWe condensate pumps take the deaerated condensate from the condenser hotwell and deliver it through the condensate demineralizer and through two strings of three low pressure feedwater heaters to a direct contact feedwater heater, which provides the equivalent performance of forward pumped heater drains. The direct contact feedwater heater receives condensate from the 5th and 6th stage feedwater heater drains, moisture separator drains and extraction steam from the LP turbines. The direct contact feedwater heater contains approximately 4 minutes of full power inventory which is required to meet ESBWR transient feedwater flow requirements.

There are four 4.8 MWe feedwater booster pumps which take suction from the direct contact feedwater heater. Feedwater booster pumps are sized to supply normal feedwater pump suction pressure. The feedwater booster pump discharges directly to a corresponding feedwater pump suction. There are four 9.3 MWe variable speed induction motor driven feedwater pumps. The reactor feedwater pumps discharge through two high pressure feedwater heaters to the RPV.

4.6.3.3 Auxiliary systems

The radioactive waste management system consists of liquid, solid, detergent, and laundry waste management, and mobile systems. Liquid waste processing is done on a batch basis. Equip-

ment drains and other low conductivity wastes are treated by filtration, UV/ozone, demineralization and transferred to the condensate storage tank for reuse. Floor drains and other high conductivity wastes are treated by filtration and ion exchange prior to being either discharged or recycled for reuse. Detergent wastes of low activity are treated by filtration, sampled and released via the liquid discharge pathway. Chemical wastes are treated by filtration, sampled and released from the plant on a batch basis. Connections are provided for mobile processing systems that could be brought in to augment the installed waste processing capability.

Connections for addition of a permanent evaporation subsystem are provided in the event that site conditions warrant. Mixed waste will be segregated from the other types of radioactive waste for packaging.

The wet solid waste processing subsystem consists of a built in dewatering station. A high integrity container (HIC) is filled with either sludge from the phase separator or bead resin from the spent resin tanks. Spent cartridge filters may also be placed in the HIC.

Dry wastes consists of air filters, miscellaneous paper, rags, etc., from contaminated areas; contaminated clothing, tools and equipment parts that cannot be effectively decontaminated; and solid laboratory wastes. The activity of much of this waste is low enough to permit handling by contact. These wastes are collected in containers located in appropriate areas throughout the plant. The filled containers are sealed and moved to an enclosed , access controlled area for temporary storage.

Connections are provided for mobile processing systems that could be brought in to augment the installed waste processing capability.

The turbine component cooling water system (TCCW) cools non-safety related components in the turbine building and rejects heat to the PSWS. The TCCW is a single loop system consisting of two 100% capacity pumps and heat exchangers.

4.6.4 Instrumentation and control systems

4.6.4.1 Design concepts including control room

The ESBWR control and instrument systems provide manual and automatic means to control plant operations and initiate protective actions should plant upset conditions occur. The ESBWR utilizes digital controllers, interfacing with plant equipment, sensors and operator controls through a multiplexing system for signal transmission to achieve these functions. The key distinguishing simplification features for plant control and monitoring include:

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

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

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

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

The MCR panels and other MCR operator interfaces are designed to provide the operator with information and controls needed to safely operate the plant in all operating modes, including startup, power operation, refuelling, shutdown, and cold shutdown. Human factors engineering principles have been incorporated into all aspects of the ESBWR MCR design.

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

4.6.4.2 Reactor protection system and other safety systems

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

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

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

The standby liquid control system (SLCS) provides an alternate method of reactor shutdown from full power to cold subcritical by the injection of a neutron absorbing solution to the RPV. SLCS initiates automatically as required to mitigate an anticipated transient without scram (ATWS).

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

The neutron monitoring system (NMS) provides indication of neutron flux in the core in all modes of reactor operation. The safety related NMS functions are the startup range neutron monitor (SRNM), the local power range monitor (LPRM), and the average power range monitor (APRM). The non safety related subsystem is the automated fixed in core probe (AFIP). The LPRMs and APRMs make up the power range neutron monitor (PRNM) subsystem. The NMS provides signals to the RPS, the rod control and information system (RC&IS), and the process computer system. The NMS provides trip signals to the RPS to scram the reactor on high neutron flux or high thermal power. In the startup range, the NMS provides a trip signal that prevents fuel damage resulting from abnormal transients in this power range. In the startup range, the SRNM provides a trip signal for excessively short reactor periods to mitigate neutron flux excursions.

4.6.5 Electrical systems

4.6.5.1 Operational power supply systems

On-site power is supplied from either the plant turbine generator, utility power grid, or an off-site power source depending on the plant operating status. During normal operation, plant loads are supplied from the main generator through the unit auxiliary transformers. A generator breaker allows

the unit auxiliary transformers to stay connected to the grid to supply loads by back feeding from the switchyard when the turbine is not on-line.

Individual voltage regulating transformers supply 120 V AC non-safety related control and instrument power.

Standby AC power supply

The non-safety related Standby AC power supply consists of two diesel generators (DG) or combustion gas turbines (CGT). Each unit provides 6.6 KV AC power to one of the two load groups whenever the main turbine generator and the normal preferred off-site power source are not operating. When operating, the standby AC power supply provides power to non-safety related investment protection loads but can be connected to power safety related loads. In ESBWR all Class 1E loads are supplied power by eight Class 1E 125 V DC batteries and eight Class 1E inverters. Other non-safety related loads are not powered from the standby power source.

Direct current power supply

Non-Class 1E DC power is supplied through four non-Class 1E 600 V AC motor control centers (MCCs). Each of the two load groups receives power from two of the non-Class 1E MCCs. One MCC in each group provides power to a 250 V DC bus through a battery charger. A 250 V DC station battery provides backup to the supply from the battery charger. A 250 V DC station battery provides backup to the supply from the battery charger. The 250 V DC batteries supply the DC motors that protect rotating machinery in case of plant power loss and supply the large inverters that power the plant's normal C & I loads.

During a loss of off-site power, the non-Class 1E systems are powered automatically from the standby diesel generators or combustion gas turbines. If these are not available, power to essential loads is provided by the 125 V DC and 250 V DC station batteries.

Instrument and control power supply

The instrument and control power supply provides 120 V AC single phase power to instrument and control loads that do not require an uninterruptable power source.

4.6.5.2 Safety related systems

Direct current power supply

The Class 1E DC power supply provides power to the Class 1E vital AC buses through inverters, and to 125 V DC loads required for safe shutdown. Each of the four divisions of class 1E DC power is separate and independent. Each division has a 125 V DC battery and a battery charger fed from its divisional 600 V AC MCC. This system is designed so that no single failure in any division of the 125 V DC system will prevent safe shutdown of the plant.

During a total loss of off-site power, the Class 1E system is powered automatically from two non-Class 1E standby diesel generators or combustion gas turbines. If these are not available, each division of Class 1E isolates itself from the non-Class 1E system, and power to safety related loads is provided uninterrupted by the Class 1E batteries. The batteries are sized to power safety related loads for a 72 hour period.

Vital (uninterruptable) power supply

The Class 1E vital AC power supply provides redundant, reliable power to the safety logic and control functions during normal, upset and accident conditions.

Each of the four divisions of this class 1E vital AC power is separate and independent. Each division is powered from an inverter supplied from a Class 1E DC bus. The DC bus receives its power from a divisional battery charger and battery. Provision is made for automatic switching to an alternate Class 1E non vital supply in case of failure of the inverter.

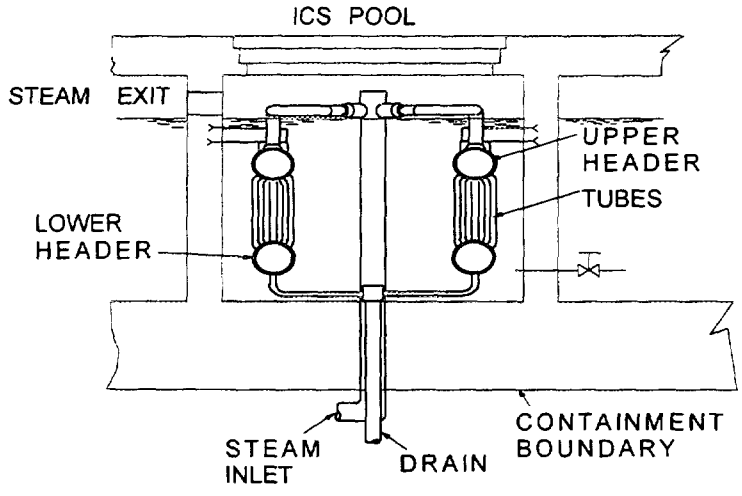


Figure 4.6-4 Isolation condenser arrangement

4.6.6 Safety concept

4.6.6.1 Safety requirements and design philosophy

The basic ESBWR safety design philosophy is built on utilization of inherent margins (e.g., larger volumes and water inventory) to eliminate system challenges. An example of this philosophy is that during a reactor isolation, no SRV shall actuate. The first line of defense is to enhance the normal operating systems ability to handle transients and accidents through such design features as adjustable speed motor driven feedwater pumps, and higher capacity CRD pumps with backup power (6.6 kV Plant investment protection buses). As a second line of defense, passive safety related systems have been included in the design to provide confidence in the plant's ability to handle transients and accidents. The plant also retains several motor driven (non safety) systems to handle transients and accidents. Also, safety related systems are designed such that no operator actions are needed for 72 hours following a design basis accident to maintain safe stable conditions. Operator actions are necessary to maintain main control room (MCR) habitability. A description of some important passive safety related systems is provided in the following section.

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

Isolation condenser system (ICS)

The isolation condenser system removes decay heat after any reactor isolation during power operations. Decay heat removal limits further pressure rises and keeps the RPV pressure below the SRV pressure set point. The ICS consists of four independent loops, each containing a heat exchanger that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC pool which is vented to the atmosphere. The arrangement of the IC heat exchanger is shown in Fig. 4.6-4.

The ICS is initiated automatically by any of the following signals: high reactor pressure, MSIV closure, or an RPV level 2 signal. To start an IC into operation, the IC condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam water interface in the IC tube bundle moves downward below the lower headers. The ICS can also be initiated manually by the operator from the MCR by opening the IC condensate return value.

The IC pool has an installed capacity that provides at least 72 hours of reactor decay heat removal capability. The heat rejection process can be continued indefinitely by replenishing the IC pool inventory. The ICS passively removes sensible and core decay heat from the reactor when the normal heat removal system is unavailable. Heat transfer from the IC tubes to the surrounding IC pool water is accomplished by natural convection, and no forced circulation equipment is required.

Emergency core cooling - gravity driven cooling system (GDCS)

Emergency core cooling is provided by the GDCS in conjunction with the automatic depressurization system (ADS) in case of a LOCA. When a Level 1 signal is received, the ADS will depressurize the reactor vessel and the GDCS will inject sufficient cooling water to keep the core covered. In the event of a postulated severe accident that results in a core melt with the molten core reaching the lower drywell region, GDCS will flood the lower drywell cavity region with the water inventory of the GDCS pools. The GDCS is completely automatic in actuation and operation. A backup to automatic actuation is the ability to actuate the system manually.

The GDCS is composed of four identical and completely independent divisions which take suction from 3 independent GDCS pools. Each of the four GDCS divisions are electrically and mechanically separated.

A confirmed RPV Level 1 signal will actuate the ADS to reduce RPV pressure. An RPV Level 1 signal initiates GDCS logic and starts two time delays. The first time delay (150 seconds), when timed out, actuates "squibb" valves in each of the four divisions providing an open flow path from the GDCS pools to the RPV. When the second time delay times out (30 minutes.) and when RPV level reaches 1 m (3.28ft) above the top of the active fuel, "squibb" valves are actuated in the four GDCS divisions that opens a flow path from the suppression pool to the RPV.

Passive containment cooling system (PCCS)

The PCCS is a passive system which maintains the containment within its pressure limits for design basis accidents such as a LOCA. It consists of four low pressure, totally independent, loops, each containing a steam condenser in a pool of water (Figure 4.6-5).

The steam condenser condenses steam on the tube side and transfers heat to the water in the IC/PCC pool. The IC/PCC pool is vented to the atmosphere. Each PCCS condenser is located in a subcompartment of the IC/PCC pool, and all pool subcompartments communicate at their lower elevations. This allows full use of the collective water inventory, independent of the operational status of any given PCCS loop.

The PCCS loops are driven by the pressure difference created between the containment drywell and the suppression pool during a LOCA. PCCS operation requires no sensing, control, logic or power actuated devices for operation. Together with the pressure suppression containment system, the four PCCS condensers limit containment pressure to less than its design pressure for at least 72 hours after a LOCA without inventory makeup to the IC/PCC pool.

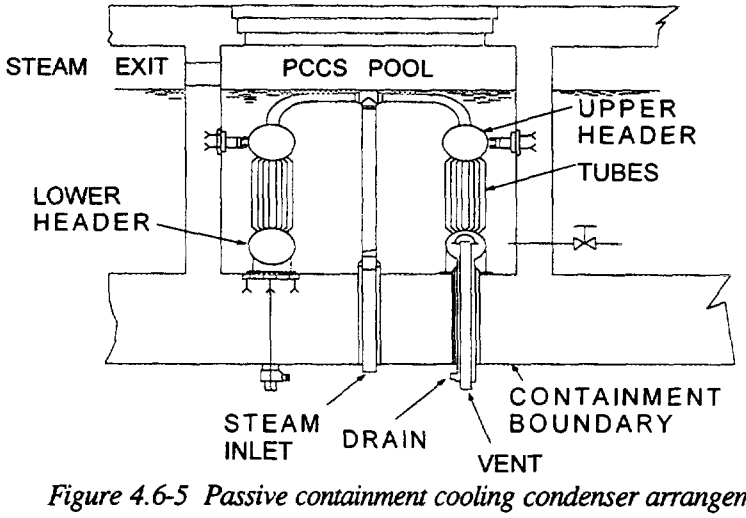


Figure 4.6-5 Passive containment cooling condenser arrangement

Table 4.6-I Key ESBWR features for prevention and mitigation of severe accidents

Design Feature	Purpose/Description
Compact containment design with minimum penetrations. Lower drywell kept dry (Mitigation)	Containment isolation with minimum leakage. High retention of aerosols. Fuel coolant interactions minimized.
Isolation condenser system (Prevention)	Controls reactor pressure. First line of defense against accidents.
Automatic depressurization system (Prevention)	Depressurizes reactor pressure vessel and prevents high pressure core melt. Minimizes probability of direct containment heating.
Passive containment cooling system (Prevention and Mitigation)	Provides long term containment cooling. Keeps pressure within design limits.
PCC heat exchanges (Mitigation)	Filter aerosols - minimize offsite dose.
Suppression pool and airspace (Prevention & Mitigation)	Suppression pool is heat sink. Scrubs aerosols. Airspace volume is sized for 100% metal water reaction.
GDCS in wetwell configuration (Prevention & Mitigation)	Increases airspace volume to handle non-condensable gas release in severe accident situations.
Core catcher (Mitigation)	Retention of molten core. Core catcher prevents basemat erosion and melt through. Prevents core-concrete interaction.
Lower drywell configuration (Mitigation)	Lower drywell floor provides sufficient spreading area ($0.04m^2$/MWt) for cooling of molten core.
Flooder system (Mitigation)	Provides additional cooling for corium on the floor.
Inerted containment (Prevention & Mitigation)	Prevents hydrogen detonation.
Recombiners / igniters (Prevention & Mitigation)	Prevents hydrogen and/or oxygen combustion and detonation.
Containment overpressure protection system (Mitigation)	An optional system that provides additional defense in depth.

The PCCS condensers are a closed loop extension of the containment pressure boundary and are designed for twice the containment design pressure. There are no containment isolation valves and they are always in "ready standby".

4.6.6.3 Severe accidents (Beyond design basis accidents)

The ESBWR design philosopy on plant safety is one of "prevention and mitigation through simplification". Prevention is achieved by utilizing a systematic design approach that provides simplified but diverse and redundant systems or components. Mitigation is achieved in two ways. First, by assuring the integrity of the containment under severe accident conditions. Second, by providing adequate fission product control so as to minimize offsite dose and consequences to the general popullation. Key ESBWR features with respect to prevention and mitigation of severe accidents are summarized in Table 4.6-I, and a proposed method of providing a desired corium spreading area is shown in Figure 4.6-6.

4.6.7 Plant layout

4.6.7.1 Buildings and structures, including plot plan

The plot plan, showing the general layout of the ESBWR buildings, is depicted in Fig. 4.6-7. The principal plant structures of the ESBWR are: the reactor building, the control building, the turbine building, the radwaste building, and the electrical building.

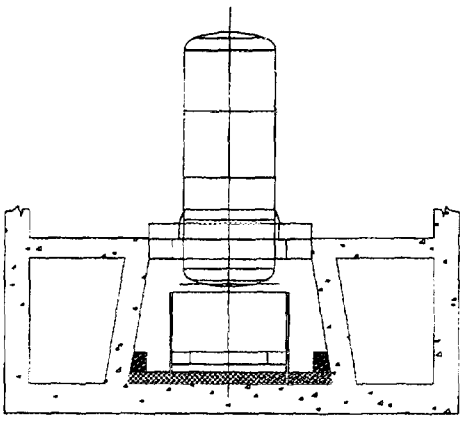


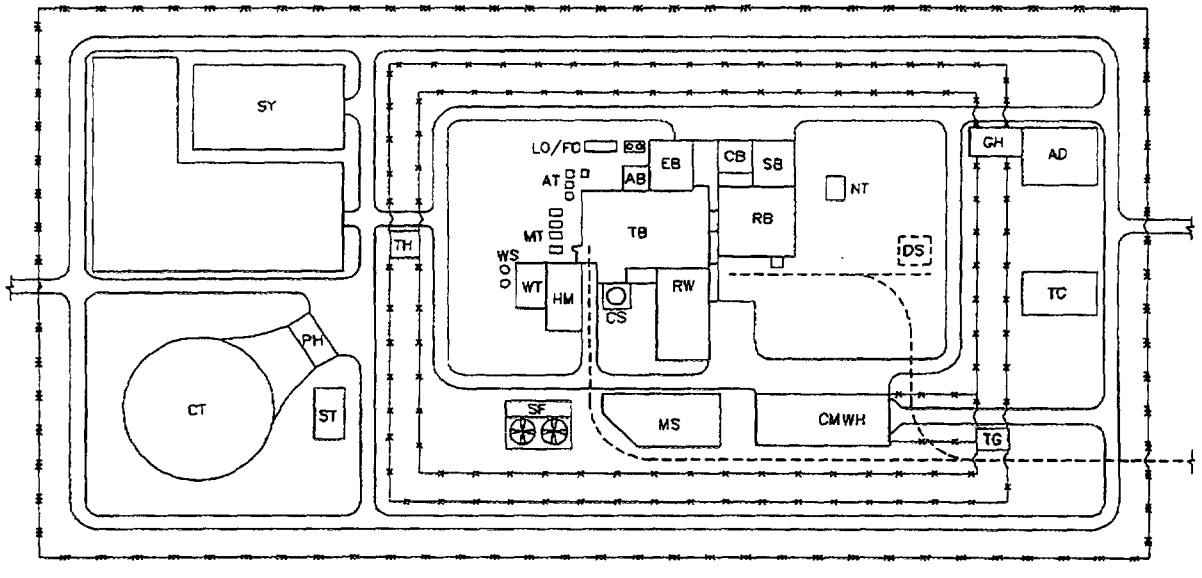
Figure 4.6-6 Corium spreadability

Design requirements

Earthquake - The reactor building seismic design is based on the US NRC Regulatory Guide 1.60 spectrum for a wide range of soil conditions. European seismic requirements will be factored in the ESBWR design.

Aircraft crash - The reactor building will be designed to the requirements of the EUR.

Explosion pressure wave - The reactor building will be designed to the requirements of the EUR.



BUILDING LEGEND:

AB	= AUXILIARY BOILER	NT	= NITROGEN STORAGE TANK
AD	= ADMINSTRATION BUILDING	PH	= PUMP HOUSE
AT	= UNIT AUXILIARY TRANSFORMER	RB	= REACTOR BUILDING
CB	= CONTROL BUILDING	RW	= RADWASTE BUILDING
CM	= COLD MACHINE SHOP	SB	= SERVICE BUILDING
CS	= CONDENSATE STORAGE TANK	SF	= SERVICE WATER & FIRE BUILDING
CT	= MAIN COOLING TOWER	ST	= SEWAGE TREATMENT PLANT
DS	= FUTURE DRY FUEL STORAGE	SY	= SWITCH YARD
EB	= ELECTRIC BUILDING	TB	= TURBINE BUILDING
GH	= GUARD HOUSE	TC	= TRAINING CENTER
HM	= HOT MACHINE SHOP & STORAGE	TG	= TEMPORARY GUARD HOUSE (W)
LO/FO	= LUBE OIL / FUEL OIL	TH	= TEMPORARY GUARD HOUSE (E)
MS	= MISCELANEOUS SERVICE AREA	WH	= WAREHOUSE
MT	= MAIN TRANSFORMER	WS	= WATER STORAGE
		WT	= WATER TREATMENT

Figure 4.6-7 ESBWR Plant Layout, Site Plot Plan

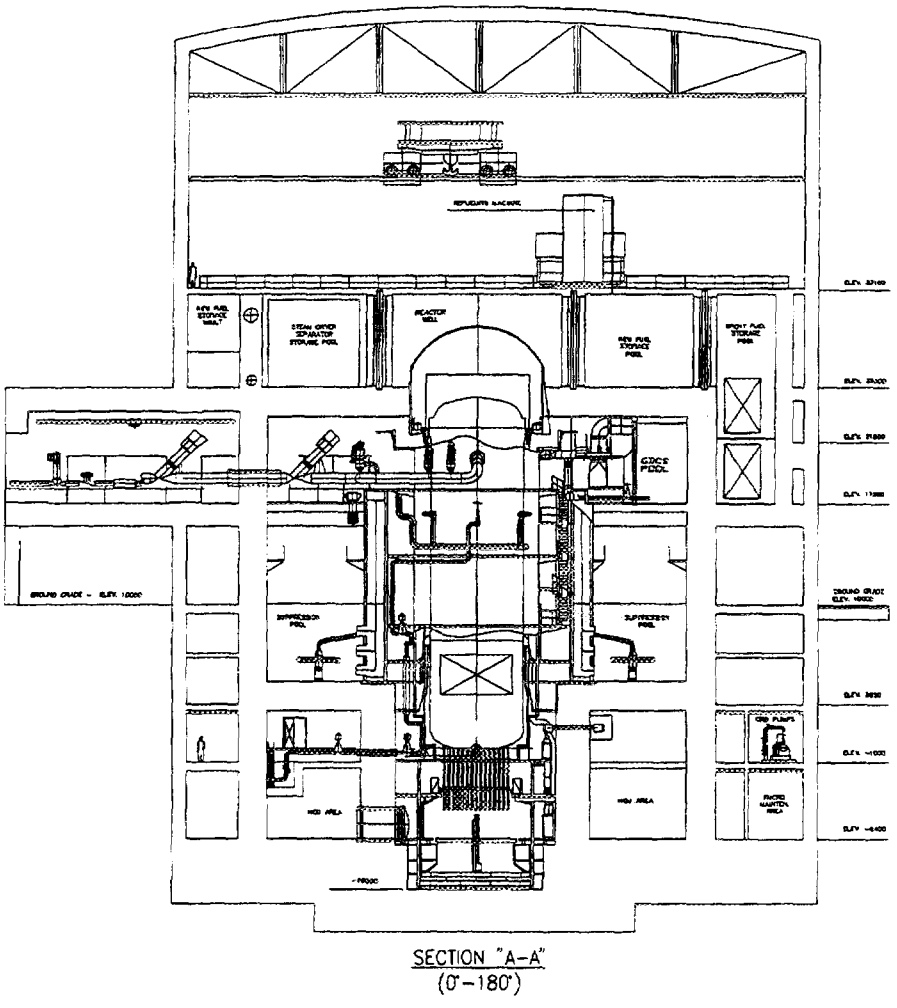


Figure 4.6-8 ESBWR Reactor Building Arrangement, Section A-A, 0° - 180°

Internal hazards - Internal loads from hazards will be incluced in the containment design. Pipe whip design will be minimized by the use of leak before break design.

Physical separation aspects - Safety grade systems are protected by physical, or spatial, separation wherever possible. When spatial separation is not possible, physical barriers will be used to provide the required equivalent separation. EUR requirements will be incorporated into the ESBWR design.

Radiation protection aspects (accessibility, shielding, ventilation) - The reactor building layout has separated the controlled from the non-controlled areas through the separation of the respective equipment by floor assignment.

4.6.7.2 Reactor building

Most of the components, equipment and systems providing safety related functions in the ESBWR are housed in the reactor building, the main steam tunnel, and pools located beneath the operating floor. This includes the reactor containment, the refuelling area with spent fuel storage, reactor water cleanup and shutdown cooling system and support equipment; non-safety related systems are surrounding this envelope. Figures 4.6-8 and -9 & -10 show the major reactor building layout.

The reactor building is a Seismic Category I structure. Its structure is integrated with that of a stepped cylindrical concrete containment vessel (RCCV), located on a common basemat. The reactor building outer walls are reinforced concrete shear walls. The building is partially embedded.

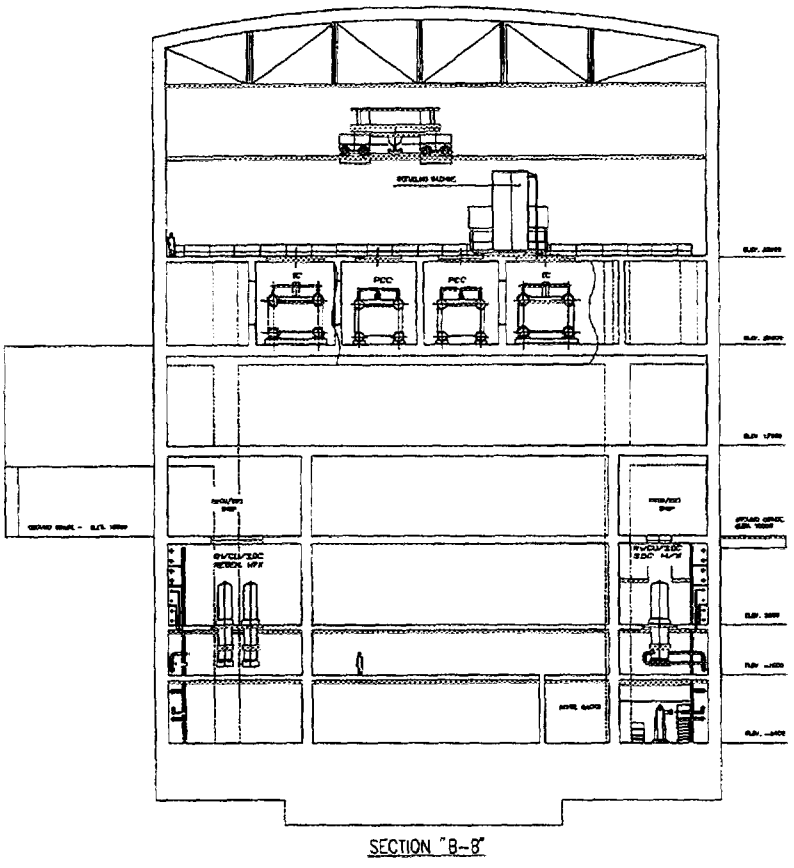


Figure 4.6-9 ESBWR Reactor Building Arrangement, Section B-B, 90° - 270°

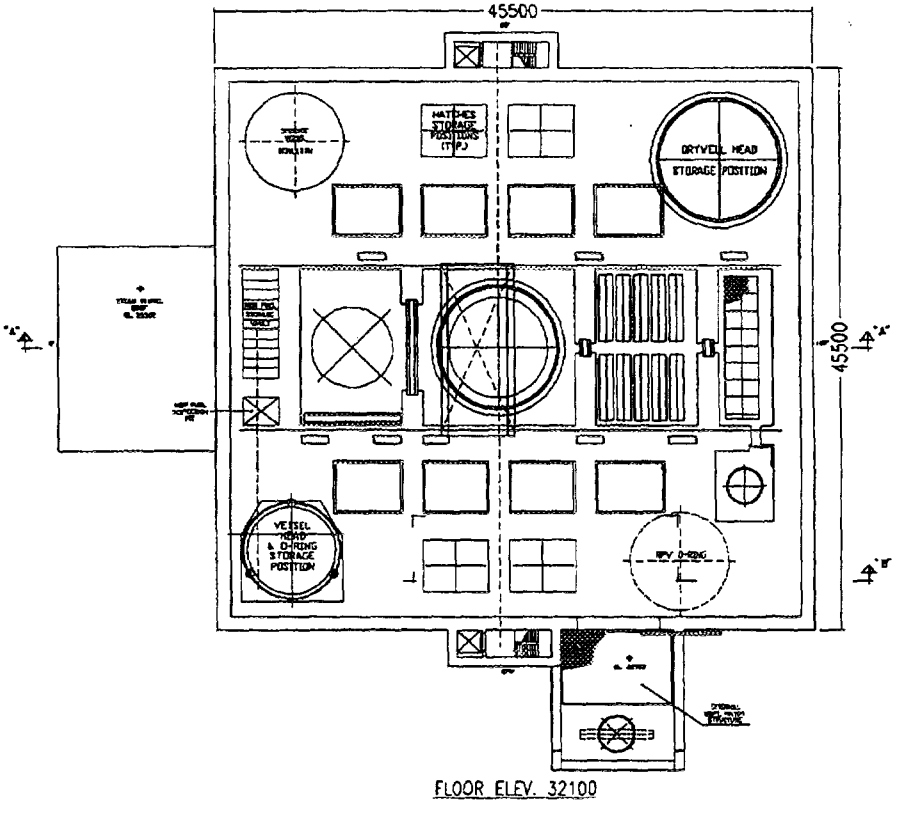


Figure 4.6-10 ESBWR Reactor Building Arrangement, Operating Floor, Elev. 32100 mm

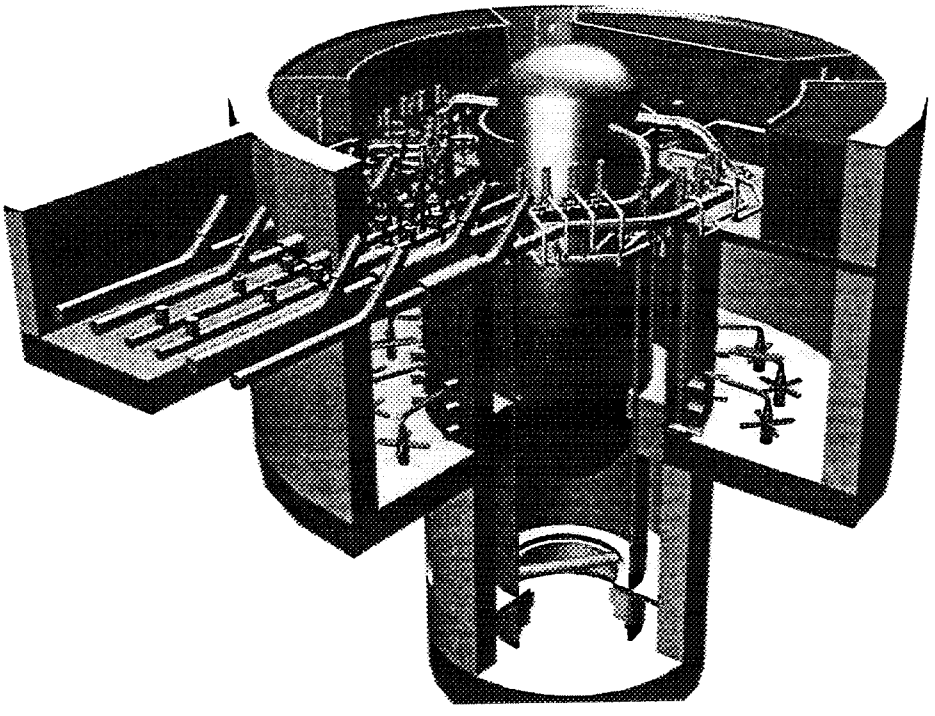


Figure 4.6-11 ESBWR Containment Layout

4.6.7.3 Containment

The ESBWR containment structure is a reinforced concrete cylindrical structure which encloses the reactor pressure vessel (RPV) and its related systems and components. The containment is divided into a drywell region and a suppression chamber region with interconnecting vent system.

The drywell region is a leaktight gas space surrounding the RPV and reactor coolant pressure boundary. It provides confinement of radioactive fission products, steam and water released in the unlikely event of a LOCA. The containment is designed to direct the fission products, steam and water released during a LOCA to the suppression pool via the vent system.

The suppression chamber region consists of the suppression pool and the gas space above it. The suppression pool is a large body of water which absorbs the LOCA energy by condensing steam from safety/relief valve discharges and RPV blowdown energy.

The gas space above the suppression pool is leak tight and sized to collect and retain the drywell gases following a pipe break in the drywell, without exceeding the containment design pressure. Figure 4.6-11 shows the containment layout.

4.6.7.4 Turbine building

The turbine building - houses equipment associated with the main turbine and generator and their auxiliary systems and equipment including the condensate purification system and the process offgas treatment system. In addition, the fuel and auxiliary pool cooling (F&APCS) system and the reactor component cooling water (RCCW) system are located in the turbine building. The turbine building is a reinforced concrete structure up to the turbine operating deck. Above the operating deck the turbine building is constructed of steel frame and metal siding. It is built at grade elevation. Shielding is provided for the turbine on the operating deck. The turbine generator and condenser are supported on spring type foundations. The turbine building is a non-safety related structure.

4.6.7.5 Other buildings

Control building

The control building houses the main control room, electrical distribution, and safety grade batteries. The lower floors of the control building, located below grade, house the safety grade equipment and are designed and built to Seismic Category I requirements. The upper floors housing the main control room and electrical distribution equipment a non safety grade structure but designed as a Seismic Category I structure.

Radwaste building

The radwaste building houses equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant. The structure up to grade elevation is reinforced concrete (first story), and has a structural steel framework with metal siding and a metal roof above that. The reinforced concrete portion of radwaste building below grade is designed to the requirements of Regulatory Guide 1.143, and the balance of the structure is classified non-seismic.

Electrical building

The electrical building houses the two non-safety related standby diesel generators (or combustion gas turbines) and their associated auxiliary equipment.

4.6.8 Technical data

General plant data		
Power plant output, gross		MWe
Power plant output, net	1190	MWe
Reactor thermal output	3613	MWt
Power plant efficiency, net	34	%
Cooling water temperature		°C
Nuclear steam supply system		
Number of coolant loops	1	
Reactor pressure vessel free volume	867	m^3
Steam flow rate at nominal conditions [7030 t/h]	1952.8	kg/s
Feedwater flow rate at nominal conditions	1952.8	kg/s
Reactor coolant system		
Primary coolant flow rate	12,940	kg/s
Reactor operating pressure	7.17	MPa
Steam temperature/pressure	287.7/7.17	°C/MPa
Feedwater temperature	214.6	°C
Core coolant inlet temperature	277.5	°C
Core coolant outlet temperature	287.8	°C
Mean temperature rise across core	10.2	°C
Reactor core		
Active core height	2.743	m
Equivalent core diameter	4.73	m
Heat transfer surface in the core	4,653	m^2
Average linear heat rate	16.6	kW/m
Fuel weight	109	t U
Average fuel power density	18.35	kW/kg U
Average core power density	48.0	kW/l
Thermal heat flux, F_q	430	kW/m^2
Enthalpy rise, F_{Π}	(later)	

Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	2,743	mm
Rod array lattice	8×8 [or 9×9] in square	
Number of fuel assemblies	1132	
Number of fuel rods/assembly	60	
Number of spacers	(later)	
Enrichment (range) of first core, average	3.95	Wt%
Enrichment of reload fuel at equilibrium core	(later)	Wt%
Operating cycle length (fuel cycle length)	24	months
Average discharge burnup of fuel	34.6	GWd/*t*
Cladding tube material	annealed, recrystallised Zr 2	
Cladding tube wall thickness	0.8128	mm
Outer diameter of fuel rods	12.3	mm
Fuel channel/box; material	Zr-4	
Overall weight of assembly, including box	(later)	kg
Uranium weight/assembly	(later)	kg
Active length of fuel rods	2,743	mm
Burnable absorber, strategy/material	Gd_2O_3 mixed with fuel	
Number of control rods	269	
Absorber material	B_4C	
Drive mechanism	electro-mechanical	
Positioning rate 30	mm/*s*	
Soluble neutron absorber	Sodium Pentaborate	
Reactor pressure vessel		
Inner diameter of cylindrical shell	7,100	mm
Wall thickness of cylindrical shell	182	mm
Total height, inside	25,400	mm
Base material: cylindrical shell	low-alloy carbon steel	
RPV head	[to ASTM A533, grade B, ASTM A508, class 3, or equiv.]	
lining	Ni-Cr-Fe alloy cladding	
Design pressure/temperature	8.62/300	MPa/°C

Transport weight (lower part) [includ. head]	781	t
RPV head		t

Reactor recirculation pump

Not applicable

Type	
Number	
Design pressure/temperature	MPa/°C
Design mass flow rate (at operating conditions)	kg/s
Pump head	MPa
Rated power of pump motor (nominal flow rate)	kW
Pump casing material	
Pump speed (at rated conditions)	rpm
Pump inertia	kg m^2

Primary containment

Type	Single wall, Pressure suppression	
Overall form (spherical/cyl.)	cylindrical, reinforced concrete with steel liner	
Dimensions (diameter/height): lower drywell	13.2/12	m
wetwell & upper drywell	33/24.6	m
Design pressure/temperature	483/171	kPa(abs)/°C
Design leakage rate	0.5	vol%/day
Is secondary containment provided?	yes	

Reactor auxiliary systems

Reactor water cleanup/Shutdown cooling,	capacity	.0011	kg/s
	filter type	mixed bed type	
Residual heat removal,	at high pressure	132.3	kg/s
	at low pressure	132.3	kg/s
Coolant injection,	at high pressure	NA	kg/s
	at low pressure	NA	kg/s

Power supply systems

Main transformer,	# of units	three 1phase	475	MVA
		one spare	475	MVA
	rated capacity		1425	MVA
Unit transformers,	# of units	two	70	MVA
			dual voltage secondaries	
Reserve transformers		two	40	kVA
			dual voltage secondaries	
Medium voltage busbars four		11.5 or 10	kV	
		four	6.6 or 6.0	kV
Number of low voltage busbar systems		four	600 vac 1E	
		four	600 vac, turbine building	
		two	600 vac reactor building	
		two	600 vac conrol building	
		two	600 vac radwaste building	
		two	600 vac mechanical draft cooling tower	
		two	600 vac natural draft cooling tower	
Standby diesel generating units		two	6.6 or 6.0	kV
			6.8/8	MWe/MVA
Number of diesel backed busbar systems			four 6.6 or 6.0 kV bussbars	
Number of DC distributions		eight	125 VDC 1E four normal 2 hour, our 72 hour coping)	
		two	125 V DC non 1E	
		two	250 V DC non 1E	
Number of battery backed busbar systems				
		eight	220 V AC 1E four normal 2 hour, our 72 hour coping)	
		two	380/220 V AC non 1E	

Turbine plant

Number of turbines per reactor	one	
Type of turbine(s)	tandem compound, four flow	
Number of turbine sections per unit (e.g. HP/LP/LP)	one HP / two LP	
last stage blade length (LSB), nominal	1320 - 1525	mm
Turbine speed	1500	rpm
Overall length of turbine unit	later	m
Overall width of turbine unit	later	m
HP inlet pressure/temperature	6.8 / 287	MPa/°C

Generator

Type	4-pole, 3-phase, turbo-generator	
Rated power gross/net	1240 / 1190	MWe
Active power	1475	MVA
Voltage, nominal	24 - 27	kV
Frequency	50	Hz
Total generator mass, including exciter	later	t
Overall length of generator	later	m

Condenser

Type	series arrangement	
	two shells / four waterboxes	
Heat sink	natural draft cooling tower	
Number/type of tubes	later / titanium or stainless steel	
Heat transfer area, nominal	50,000	m^2
Cooling water flow rate, nominal		28 m^3/s
Cooling water temperature (max)	35	°C
Condenser pressure, nominal		
LP cond / HP cond	7.6 / 9.3	kPa

Condensate pumps

Number	4	
Flow rate	0.48	kg/s
Pump head	1.95	MPa
Temperature	4.68	°C
Pump speed	(later)	rpm

Condensate clean-up system

Full flow/part flow	full condensate flow
Filter type	rod type

Direct contact feedwater heater tank

Volume	500	m^3
Pressure/temperature	0.552/156	MPa/°C

Feedwater booster pumps

Number	4	
Flow rate	0.56	kg/s
Pump head	4.0	MPa
Feed pump power	9.3	MW
Feedwater temperature (final)	215.5	°C
Pump speed	(later)	rpm

Condensate and feedwater heaters

Number of heating stages,	low pressure	3
	high pressure	2

4.6.9 Project status and planned schedule

Entities involved

The ESBWR program is based on the US Department of Energy (DOE) sponsored SBWR program, that has been ongoing for about 10 years. The ESBWR program was started in 1993 to adapt the SBWR to improve economics and incorporate European requirements. A multi-year, three phase program has been defined to develop a detailed design and complete the technology over several years.

The future progress and schedule will be defined by progress on the technology and design assessment by the various partners. It is estimated that the technology and design efforts will be complete by 1999.

The design and technology program involves several utilities, designers and research groups in over seven countries. The overall design leadership is provided by General Electric Company (GE - USA) and overall program guidance is provided by an ESBWR steering committee.

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4.7 KOREA NEXT GENERATION REACTOR (KNGR), KEPCO, ETC, REPUBLIC OF KOREA

4.7.1 Introduction

The Korean Next Generation Reactor (KNGR) is a 1300 MWe PWR for which the design is based on the technological experience that has been accumulated through the development of the Korean Standard Nuclear Power Plant (KSNPP) design, a 1000 MWe PWR, and it represents an increased capacity evolution of the KSNPP design. The KNGR also incorporates a number of design modifications and improvements to meet the utility's needs for enhanced safety and economic goals and to address the new licensing issues such as mitigation of severe accidents

Since KNGR is an evolution from its predecessor, the KSNPP, the basic configuration of the nuclear steam supply system is the same, e.g., it has two steam generators with four reactor coolant pumps in a "two hot legs and four cold legs" arrangement. The KNGR has many advanced features from its predecessor, such as a double containment, direct vessel injection of emergency core cooling water, and the in-containment refuelling water supply system. The power level of the KNGR is higher yielding an electric power output of about 1300 MWe which is 30% higher than that of the KSNPP. Apart from the visible, "inherited" features, there are quite a number of design improvements also. The main control room designed with consideration of human factors, simplifications of the emergency core cooling system and the emergency feedwater system are just a few examples of the design improvements.

The reactor and plant design concept of the KNGR is based upon the results of two years of research in the Phase I of the project. During this period, the Advanced Light Water Reactor (ALWR) designs being developed by the advanced countries were reviewed and the design concepts were modified to match domestic needs and capabilities. The KNGR design concept has adopted the best features from the other ALWRs and their design criteria. The ALWRs were also reviewed quantitatively through safety and economic evaluations to establish the safety and economic goals for the KNGR. The design requirements have been established through this comparative study, and the major requirements that direct the design of the KNGR are:

General requirements

Type and capacity: PWR, 4000 MWt (core thermal power);
Plant lifetime: 60 years;
Seismic design: SSE 0.3g;
Containment building: Concrete double containment;
Safety goals: core damage frequency lower than 10E-5 per reactor year; and frequency of large radiation release due to containment failure lower than 10E-6 per year.

Performance requirements and economic goals

Plant availability: 90%;
Unplanned trips: less than 0.8 per year;
Refuelling interval: 18-24 months;
Occupational radiation exposure: less than 1 manSv per reactor and year;
Construction period: 48 months; and
Economic goal: 20% cost advantages over competitive energy sources.

As noticed, the KNGR aims at both enhanced safety and economic competitivity. From the point of view of probabilistic safety assessment (PSA), KNGR will have about 10 times lower probabilities of core damage and accidental radiation release than the KSNPP. The economic goal of KNGR to secure around 20% cost advantage over competing energy sources, such as coal-fired power generation, is considered achievable by high performance in operation and cost savings in construction.

4.7.2 Description of the nuclear systems

4.7.2.1 Primary circuit and its main characteristics

The primary loop configuration of the KNGR is similar to that of the KSNPP which has two reactor coolant loops. The nuclear steam supply system is designed to operate at a maximum core thermal output of approximately 4000 MWt to produce an electric power output of around 1350 MWe in the turbine/generator system. The major components of the primary circuit are the reactor vessel, two reactor coolant loops, each containing one hot leg, two cold legs, one steam generator (SG), and two reactor coolant pumps (RCPs), and a pressurizer (PZR) connected to one of the hot legs. All components are located inside the containment. The two SGs and the four RCPs are arranged symmetrically. The steam generators are located at a higher elevation than the reactor vessel for natural circulation purposes. For the vent and drain, the elevation of the PZR and the surge line is higher than that of reactor coolant piping. A schematic diagram of arrangements and locations of the primary components and safety-related systems are shown in Figure 4.7-1.

The design temperature in the hot leg is reduced from 327°C at the normal operating pressure, 15.5 MPa, of the currently operating nuclear plants to 324°C in order to increase the operating margin from the increased subcooledness. Because of the decreased temperature equipment corrosion problem can be reduced.

In the reactor pressure vessel (RPV) design, four direct vessel injection (DVI) lines are connected to supply emergency core cooling water from the in-containment refuelling water storage tank (IRWST). Level probes have recently been added in the hot leg to monitor the water level of the hot leg during mid-loop operation.

The capacities of the pressurizer and the steam generators (especially the secondary side) are increased from that of current designs. To smoothen plant response to relevant operating transients and accidents, the pressurizer volume is increased from 51 m^3 to 68 m^3. The increased water inventory on the secondary side smoothens normal operating transients and reduces the potential for unplanned reactor trips. In case of a total loss of feedwater supply, the postulated dryout time of the SG will be increased by more than 50% compared with current designs.

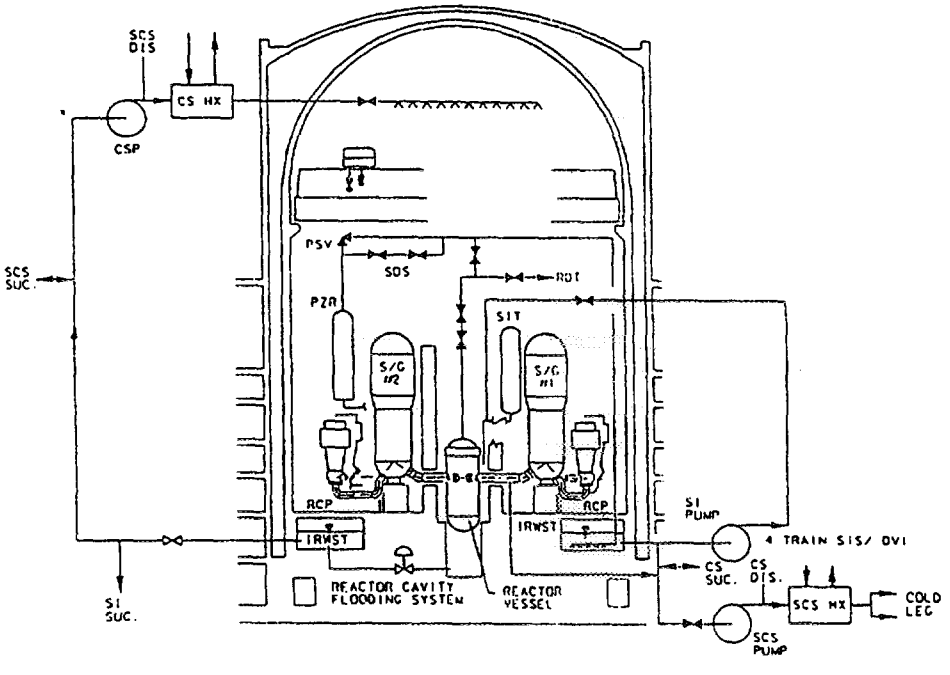


Figure 4.7-1 KNGR - Schematic diagram of primary components and safety systems

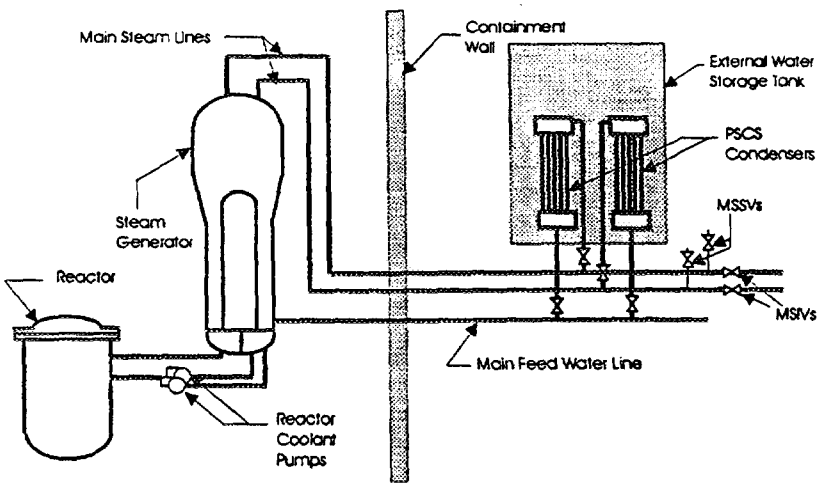


Figure 4.7-2 KNGR - Schematic diagram of PSCS

Over-pressure protection of the reactor coolant system (RCS) is provided by two-stage depressurizations through safety valves and safety depressurization valves (SDVs) which are mounted to the top of the pressurizer. Both the lines of the safety valves and the SDVs are directly connected to spargers which are submerged in the In-containment refuelling water storage tank (IRWST) to mitigate the possibility of the contamination of the containment by radioactivity release to the atmosphere. SEBIM valves are under consideration as type of safety valves.

The following safety functions can be performed through the KNGR pressurizer:

- over-pressure protection of the RCS by automatic discharge;
- depressurization of the RCS during natural circulation cooldown;
- feed and bleed cooling for decay heat removal during total loss of feedwater (TLOFW);
- prevention of core uncovery after TLOFW in condition with safety injection system (SIS);
- depressurization of the RCS for severe accidents;
- non-condensable gas venting.

On the secondary side of the SGs, two discharge trains will be arranged on each main steam line at the outlet of the SG. Each train has five non-isolatable safety valves, one relief valve (normally closed) and one isolation valve (normally open). As a new and novel means of decay heat removal in condition with the loss of feedwater event is adoption of the passive secondary condensing system (PSCS) which is located outside the containment and cools the steam. The natural circulation loop between the SG and the PSCS removes core decay heat. Figure 4.7-2 shows a schematic diagram of PSCS.

4.7.2.2 Reactor core and fuel design

The core of the KNGR is designed to generate about 4000 MWt output with an average volumetric power density of 95.9 kW/l, i.e., at a reduced density compared with the current PWR generation. The core consists of 241 fuel assemblies built up by fuel rods containing uranium dioxide fuel with an average enrichment of 2.6 w/o in a 16×16 array. Each fuel assembly consists of 236 fuel rods, 5 guide tubes. Figure 4.7-3 shows CEA and in-core instrument locations in the KNGR.

The possibility of utilizating MOX fuel is considered in the core design, and additional control element assemblies (CEAs) can be installed to increase the reactivity control capability, if necessary, when high percentage MOX fuel loadings are to be used. The number of CEAs in the basic design version is 93, with a possibility for 8 additional CEAs. About three-fourth of the CEAs are full-strength reactivity control assemblies, the rest, 25 of the 93, are part-strength CEAs.

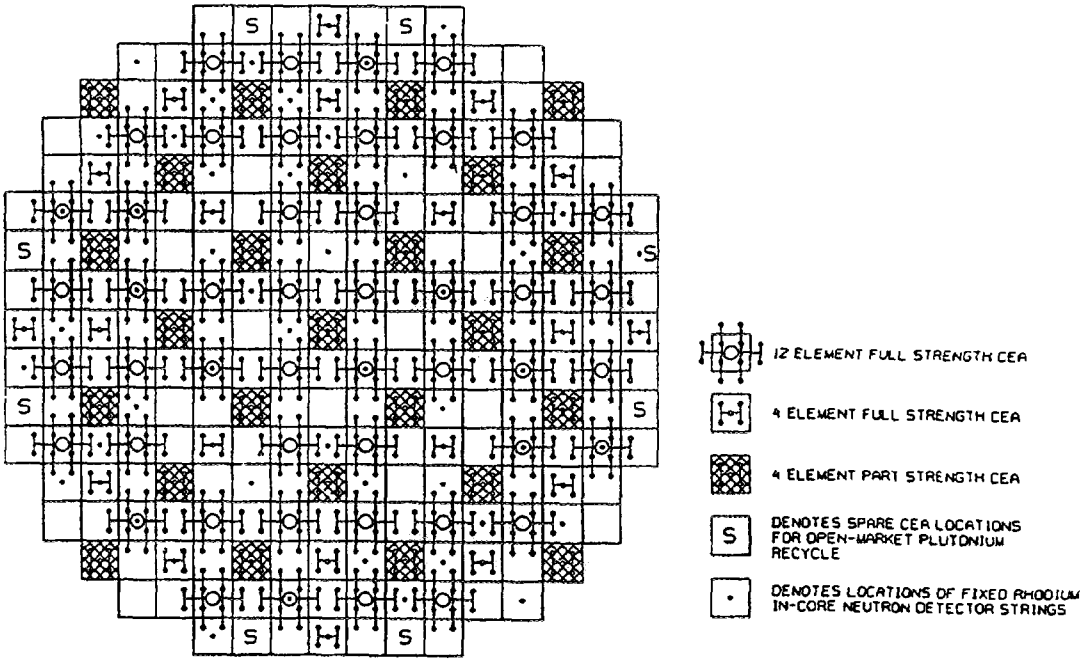


Figure 4.7-3 KNGR - Locations of CEAs and in-core instruments

The absorber materials used for full-strength control rods are boron carbide (B_4C) pellets and silver-indium-cadmium (Ag-In-Cd) bar. Inconel alloy 625 is used as the absorber material for the part-strength control rods.

The core is designed for an operating cycle of 18-24 months with a discharge burnup as high as approximately 60 000 MWD/MTU, and has an increased thermal margin of up to 15% to enhance safety and improve operation performance. This core design enhances power manoeuvering capability including daily load following and unplanned power change (10% step change, 5% ramp change/min) by improving the control rod function using 25 part strength CEAs. A portion of the fuel rods contains uranium fuel mixed with a burnable absorber (either Erbium or Gadolinium) to suppress excess reactivity after fuelling and to help control the power distribution in the core. The neutron flux shape is monitored by means of movable and fixed in-core instrumentation (ICI) assemblies.

4.7.2.3 Fuel handling and transfer systems

The fuel handling system consists of the equipment and tools for refuelling of the reactor. The system is designed for safe and rapid handling and storage of fuel assemblies from receipt of fresh fuel to shipment of spent fuel.

The major equipment of the system comprises the refuelling machine, the CEA change platform, the fuel transfer system, the fresh fuel elevator, the CEA elevator and the spent fuel handling machine. The refuelling machine is located in the containment building and moves fuel assemblies into and out of the reactor core and between the core and the fuel transfer system. The fuel transfer system rotates fuel assemblies from the vertical to the horizontal position, shuttles them through the fuel transfer tube assembly and returns them to the vertical position, either inside the containment building or in the fuel building. The spent fuel handling machine, located in the fuel building, carries fuel between the fuel transfer system, the fresh fuel elevator, the spent fuel storage racks and the spent fuel shipping cask areas. The CEA change platform, which is located in the containment building, is used to perform CEA replacement and relocation operation within the stored upper guide structure (UGS) and is used as a work platform for handling and replacement of in-core instrumentation equipment. The CEA elevator, which is located in the containment building, is used during CEA assembly replacement. The new fuel elevator which is located in the fuel

building, is used to lower fresh fuel into the spent fuel pool. A pool seal assembly is utilized to seal the annulus between the reactor vessel flange and the refuelling pool floor.

Specific tools and lift rigs are used for disassembly and handling of reactor components. Lift rigs are provided for the reactor head, core support barrel, and upper guide structure. A multiple stud tensioner is used to tension and detension the reactor head closure studs and storage racks are available for storage of the studs and nuts when they are removed from the tensioner. The upper guide structure, which consists of the fuel assembly alignment plate, control element shroud tubes, the upper guide structure base plate, CEA shrouds, and an upper guide structure support barrel, is removed from the core as a unit during refuelling by means of special lifting rig. Fuel storage racks are located in the fuel building for the storage of fresh and spent fuel.

4.7.2.4 Primary components

Reactor pressure vessel

The reactor consists of a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical closure head, internal structures, core support structures, fuel assemblies, control rod assemblies, and control and instrumentation facilities.

The major design improvements incorporated in the reactor design include: enhancement of core monitoring capability, larger operating margins, higher power level, and lower failure rate of fuel elements for higher plant availability and reliability.

The life time of the reactor pressure vessel is extended to 60 years by use of low carbon steel, which has lower contents of Cu, Ni, P, S compared to current designs, to increase brittle fracture toughness. The inner surface of the reactor vessel is clad with austenitic stainless steel or Ni-Cr-Fe alloy. In addition, it has been designed to have an end-of-life RTndt of 37.8°C (100°F). Also, the reactor pressure vessel is manufactured by the ring forging method to minimize welding parts.

Reactor internals

The reactor internals consist of the core support structures, which include the core support barrel, upper guide structure barrel assembly and lower support structure, and the internal structures. The core support structures are designed to support and orient the reactor core fuel assemblies and control element assemblies, and to direct the reactor coolant to the core. The core support barrel and the upper guide structure are supported at its upper flange from a ledge in the reactor vessel flange. The flange thickness is increased to sustain the enhanced seismic requirements.

All reactor internals are manufactured of austenitic stainless steel except for the hold-down ring which is made of high tension stainless steel. The hold-down ring absorbs vibrations caused by the load to axial direction of internal structures.

Steam generators

The steam generators are vertical U-tube heat exchangers with peerless type steam dryers, moisture separators, and an integral economizer in which heat is transferred from the reactor coolant to the main steam and feedwater system. A major improved feature incorporated into the steam generator design is the use of advanced corrosion resistant material in the steam generator tubes, namely Inconel 690 replacing Inconel 600. The steam generator water inventory is increased to provide longer forgiving time for operators in case of loss of feedwater events. In order to improve the operating margin of the steam generator, the tube plugging margin increases from 8% in the earlier designs to 10% and the time of boil-dry is also increased from 30 to 41 minutes.

Pressurizer

The pressurizer which is connected to on of the hot legs, consists of a steel pressure vessel containing pressurizer heaters, spray nozzles and safety relief valves. Its function is to maintain the pressure and water inventory of the reactor coolant system within specified limits during all normal and upset operating

conditions without actuation of pressure relief devices. The functional capability of the pressurizer is enhanced by using an increased volume relative to power and an improved pressurizer heater control manoeuvrability during reactor shutdown.

Reactor coolant pumps

The reactor coolant pumps circulate reactor coolant through the reactor vessel to the steam generators for heat removal and return it to the reactor vessel. There are two pumps for each coolant loop, located in each cold leg.

The pump is a single-stage centrifugal unit of vertical type, driven by a 12 000 hp electric motor. Leak-tightness of the shaft is ensured by a mechanical seal designed to seal against the full internal pressure in the pump.

The basic function and type of the pump in the KNGR plant is, except for the pump size and power, the same as in the Korean Standard Nuclear Power Plants.

Piping

The Leak-Before-Break (LBB) principle is adopted for the piping system of the KNGR, since the pipe whip restraint and the support of the jet impingement shield in the piping system of the earlier plant are expensive to build and maintain, and lead to a potential degradation of plant safety. The LBB technology is applicable to the main coolant lines, surge lines, and pipes of the shutdown cooling system and the safety injection system and the main steam lines in the containment. This technology reduces the redundant supports of the pipe in the NSSS pipe system. The cost of design, build-up and maintenance is reduced, since the dynamic effects of postulated ruptures in the piping system are also eliminated from the design basis.

4.7.2.5 Reactor auxiliary systems

Chemical and volume control system

The CVCS in the KNGR is designed as a non-safety-grade system to achieve a high degree of reliability by means of simplifications. Although the CVCS is not required to perform safety functions such as safe shutdown, accident mitigation, and maintaining the integrity of the reactor coolant pressure boundary, this system is essential for the normal day-to-day operation of the plant. For normal operation, only one pump will be used to supply the required minimum flow, 12.6 kg/s. The flow diagram of the chemical and volume control system is shown in Figure 4.7-4.

The operational function of the CVCS in the KNGR are as follows:

- to control chemistry and purity of the reactor coolant;
- to maintain the reactor coolant inventory;
- to recover the boron in the reactor coolant system;
- to perform auxiliary pressurizer spray;
- to supply reactor coolant pump seal injection
- to remove noble gases in the reactor coolant system; and
- to purify the secondary coolant system flow and the water in the In-containment refuelling water storage tank.

The letdown flow from the reactor coolant system passes through the regenerative and letdown heat exchanger, where an initial temperature reduction takes place via heat transfer to fluid on the shell side of the heat exchangers. Pressure reduction occurs at the letdown orifice and the letdown control valve. Following temperature and pressure reduction, the flow passes through a purification process at the filters and ion exchangers.

After passing through the purification process, the letdown flow is diverted into the volume control tank (VCT) which is designed to provide a reservoir of reactor coolant for the charging pumps and for the dedicated seal injection pumps for the reactor coolant pumps.

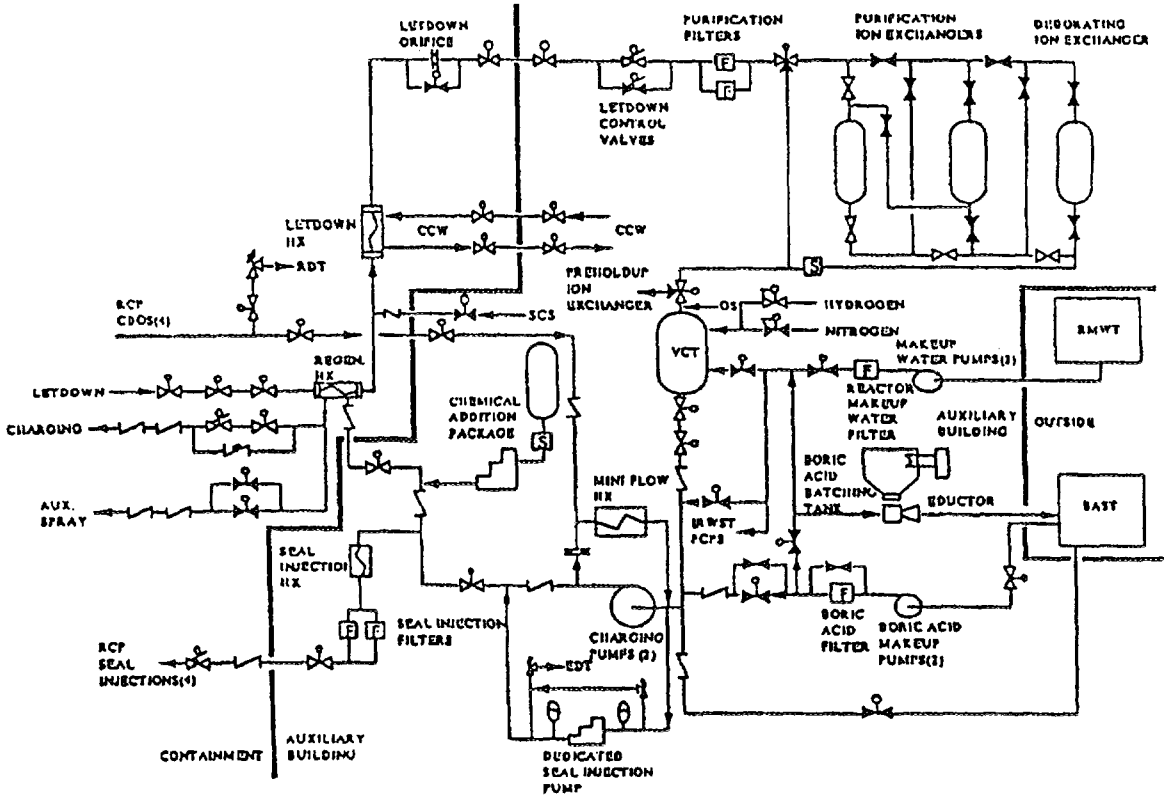


Figure 4.7-4 KNGR - Chemical and volume control system

Reactor coolant gas vent system

The RCGVS is a portion of the safety depressurization and vent system (SDVS). The reactor coolant gas vent valves are mounted to the top of the pressurizer. The size of the vent line will be increased to have sufficient capacity to vent one-half of the RCS volume in one hour assuming a single failure. Moreover, flow restricting orifices are planned to be removed from the system to improve vent capacity.

Steam generator blowdown system

The functions of the SG blowdown system are to control SG secondary side water chemistry and to remove sludge from the SG tube support plates. One flash tank can accommodate normal and high capacity blowdown flow rates. To remove dynamic loading due to two-phase flow, the flash tank for blowdown is located in the auxiliary building near by the containment. Bypass lines to the condensers are installed to overcome unavailability of the flash tank or the processing system.

Further reactor auxiliary systems

The following reactor auxiliary systems are under development:

- Primary sampling system;
- Fuel storage and handling system;
- Component cooling water system; and
- Gas, liquid and solid radwaste treatment system.

4.7.2.6 Operating characteristics

The plant power control system is capable of daily load follow operation at a typical load variation profile in Korea; 16 hours at 100% and 4 hours at 50% with 2 hours ramps for power decreases and increases. The reactor core control should be capable of a step power change of $\pm 10\%$ and ramp changes of 5% per minute without detrimental effects on the fuel rod integrity.

The load rejection capability at rated power should also be incorporated. Thus, the reactor will not automatically be tripped in the event of a turbine trip if the transient causing the turbine trip is limited to the secondary system and the primary system, including the reactor, is in the normal operation condition. This capability can reduce the outage time caused by secondary system troubles since the reactor power can be brought up to 100% as soon as the troubles have been fixed.

In addition to the above-mentioned capability, minimizing the use of soluble boron during the daily load follow operation, and enhancing the load follow capability to meet the frequency control needs of the grid are being studied.

4.7.3 Description of turbine generator plant system

4.7.3.1 Turbine generator plant

The turbine generator plant consists of main steam, extraction steam, feedwater, condensate, turbine generator and auxiliary systems. For these systems, heat balance optimization studies have been made, considering system operability, reliability, availability and economy.

The turbine generator systems are designed to be capable of operation at 3% house load for a period of at least 4 hours without any detrimental effects of the systems, and capable of startup to full load from cold conditions in 8 hours, including rotor preheat.

The main steam lines and the high pressure turbine are designed for a steam pressure of 6.9 MPa (1000 psia), and two reheater stages are provided between the high pressure and the low pressure turbines. The generator is a three phase, 4 pole unit operating at 1800 rpm.

The capacity, response and modulation capabilities of the turbine bypass system will be designed to make the turbine capable of withstanding a 100% generator load rejection without trip of the reactor or the turbine. The total flow capacity of the turbine bypass system is designed to be 55% of the turbine steam flow at full load steam pressure.

The differential pressure between the steam generator outlet nozzles is designed to be

- 6.9 kPa (1 psi) at 0-15% power operation;
- 20.7 kPa (3 psi) at 15-100% power operation; and
- less than 0.207 MPa (30 psi) for transient conditions of maximum one minute duration.

4.7.3.2 Condensate and feedwater systems

The feedwater pump configuration is selected to be 3x50% because of its ability to allow more reliable operation; all three pumps are normally operating, and the plant can remain at 100% power operation if one of the feedwater pumps is lost. On-line condensate polishers which can operate in full and partial flow, as well as in bypass mode, are provided to maintain proper water chemistry during normal power operation.

In the feedwater systems, the feedwater heaters are installed in 7 stages and arranged horizontally for easy maintenance and reliability.

4.7.3.3 Auxiliary systems

Oil filtration facilities which have on-line cleaning capability/bypass provision with a portable centrifuge, are provided for the lube oil system of the turbin.

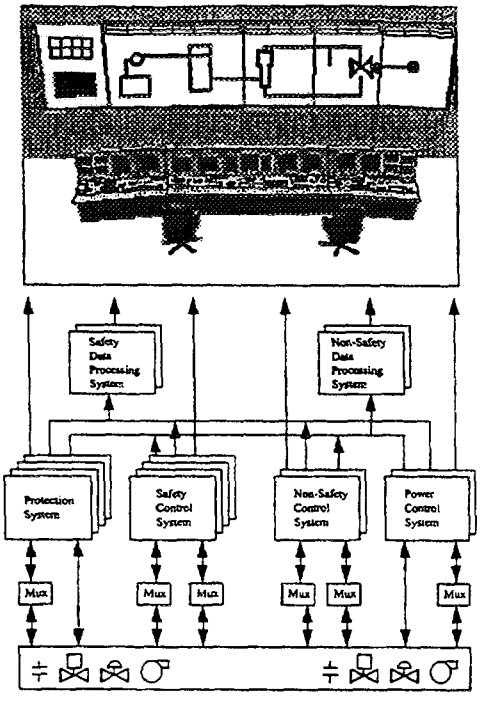


Figure 4.7-5 KNGR - Concept of MMIS configuration

4.7.4 Instrumentation and control systems

4.7.4.1 Design concepts including control room

The I&C and control room concept to be implemented in the KNGR design is schematically depicted in Figure 4.7-5.

KNGR is, like most of the advanced reactors being developed world-wide, equipped with digitized instrumentation and control (I&C) systems and computer-based control room man-machine interface (MMI), reflecting the status of modern electronics and computer technologies.

The I&C and MMI of the KNGR will be designed to be a highly reliable, safe, operable, and maintainable system which can perform the plant monitoring, control and protection functions efficiently during normal operation as well as in emergency situations by implementing advanced digital technology along with a systematic application of human factor engineering principles.

The main features of the I&C system are the use of microprocessor-based Programmable Logic Controllers (PLCs) for the control and protection systems, and the use of UNIX work stations and industrial PCs (personal computers) for data processing systems.

To protect against common mode failures in software due to the use of software-based I&C systems, different types of PLCs and computers will be required in the redundant systems for diversity. For data communication, a high speed fibre optic network based on standard protocols is used. The remote signal multiplexer is also utilized for the safety and non-safety systems field signal transmission.

The I&C system architecture will be designed with open technology to the extent possible based on industrial standards for easy upgrade and maintainability. Since the KNGR relies more on software than hardware for the I&C systems, software verification and validation for safety functions become very important. Therefore, there will be a full scope verification and validation of the software of the safety systems.

The KNGR Main Control Room (MCR) design is characterized by a reduced number of dedicated indicators and annunciators, standard unified design for all plant system MMIs, a compact operator work station, an extensive use of video display units such as ELDs [Electro Luminiscent Displays] and CRTs [Cathode Ray Tubes] for indication and annunciation, a large scale display for presentation of integrated wall mimics for overall process monitoring of the power plant, and computerized operating procedures for normal and emergency circumstances provided through video display units.

The controls in the MCR will be done primarily through video display units except for some miniature push buttons to control critical equipments for safety and reliability.

Human factor engineering is an essential element of the MCR design and the human factor engineering principles are systematically employed to ensure safe and error free operation.

4.7.4.2 Reactor protection system and other safety systems

The reactor protection system and other safety-related systems will be designed using "off-the-shelf" digital equipments which are commercially dedicated to standardize the components and minimize the maintenance cost with the consideration of diversity. A high degree of conservatism is required in the design of the safety-related systems, and therefore, design methodologies such as redundancy, diversity, and segmentation have been incorporated in order to achieve both the desired availability and reliability of these systems.

A high reliability of the protection system is ensured by self-diagnostics, and automatic functional tests through surveillance using four independent channels.

The redundant and fault tolerant configuration on controllers and the use of fiber-optics to isolate communications will increase system availability and maintainability.

A detailed software development program for software-based Class 1E systems will be produced and applied as a guideline to ensure completeness of the software implementation, verification and validation process. Several critical safety systems will be evaluated through prototyping and design verification programmes.

4.7.5 Electrical systems

The main features are of the electrical system configuration (Figure 4.7-6):

- Two independent off-site power sources of 345 kV;
- One main transformer consisting of three single-phase step-down transformers, and two three-winding unit auxiliary transformers for power delivery and supply during normal operation mode;
- Two Class 1E emergency diesel generators to provide on-site stand-by power for the Class 1E loads;
- An alternate AC source to provide power for plant equipment necessary to cope with station blackout;
- Automatic transfer of power supply from unit auxiliary transformers to standby auxiliary transformers in the event of loss of power supply through the unit auxiliary transformers;
- Four independent Class 1E 125V DC systems for each reactor protection system channel of a plant;
- Two Non-Class 1E 125V DC systems for each unit and two common Non-Class 1E 125V DC systems for the AAC and radwaste building; and
- AC voltage levels of 13.8 kV and 4.16kV for medium, 480 V and 120V for low voltages.
- Two Non-Class 1E 125V DC systems for each unit and two common Non-Class 1E 125V DC systems for the AAC and radwaste building; and
- AC voltage levels of 13.8 kV and 4.16kV for medium, 480 V and 120V for low voltages.

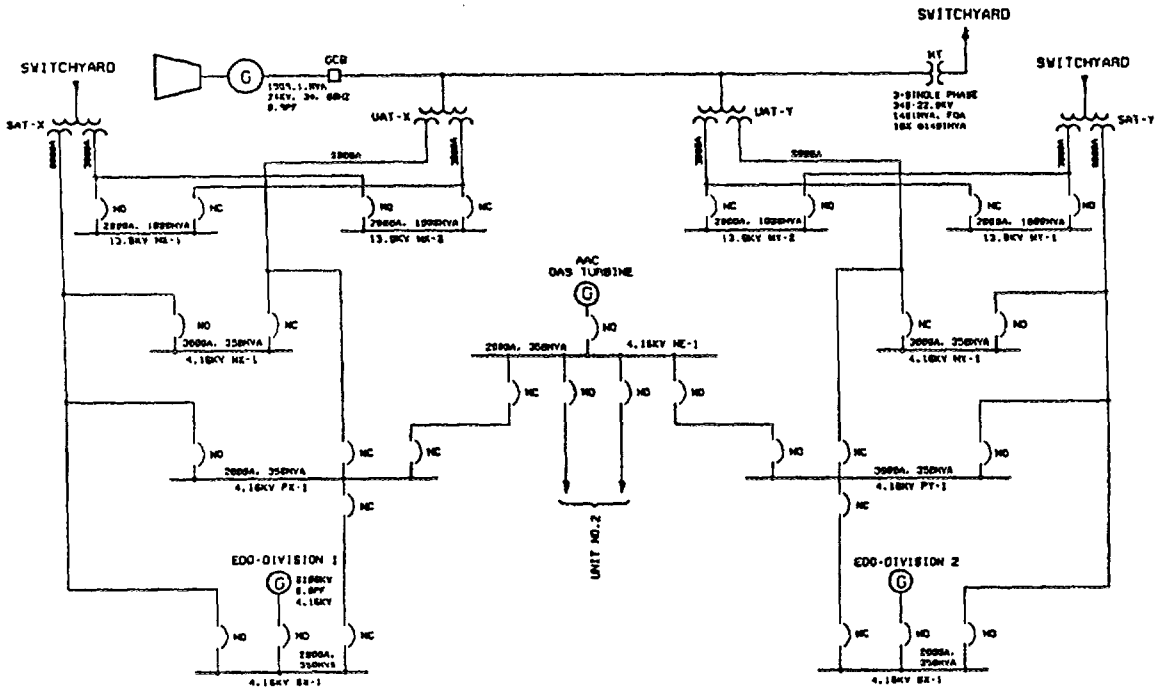


Figure 4.7-6 KNGR - Single line diagram

4.7.5.1 Operational power supply systems

The main power system consists of the generator, the generator circuit breaker, the main transformer, the unit auxiliary transformer and the stand-by transformer. The generator is connected to a gas-insulated 345 kV switchyard via the main transformer which is made up by three single-phase transformer units. Step-down unit auxiliary transformers are connected between the generator and the main transformer, and supply power to the unit equipment for plant startup, normal operation and shutdown. The stand-by transformer is always energized and ready to ensure rapid resumed power supply to the plant auxiliary equipment in the event of failure of the main and unit auxiliary transformers.

The arrangement of the on-site electrical distribution system is based on the functional characteristics of the plant equipment to ensure reliability and redundancy of power sources.

The normal power source for non-safety and permanent non-safety loads is the off-site power source and the generator. If the normal power source is not available, the permanent non-safety loads are covered by two alternative sources: one from the stand-by off-site power source (via the stand-by transformer) and the other from one non-1E alternate AC power source with a gas turbine generator.

4.7.5.2 Safety-related systems

The electric power necessary for the safety-related systems is supplied through 4 alternative ways: firstly, the normal power source, i.e., the normal off-site power and the in-house generation; secondly, the stand-by off-site power, i.e., the off-site power connected through the stand-by transformer; thirdly, the on-site standby power supply, i.e., two diesel generators; and finally, the alternative AC source, i.e., the gas turbine generator.

Among these power sources, the on-site standby power is the most crucial for safety; it should be highly available in any situation. The arrangement of the on-site electrical distribution system is based on the functional characteristics of the equipment to ensure reliability and redundancy of power sources.

The on-site power supply is ensured by two independent Class 1E diesel generator sets; each of them is located in a separated building and is connected to one 4.16 kV safety bus.

The alternate AC source adds more redundancy to the electric power supply even though it is not a safety grade system. The non-class 1E alternate AC is provided to cope with Loss-of-Off-site-Power (LOOP) and Station Blackout (SBO) situations which have a high potential of transients progressing to severe accidents. The alternate AC source is sized with sufficient capacity to accommodate the loads on the safety and the permanent non-safety buses.

4.7.6 Safety concept

4.7.6.1 Safety requirements and design philosophy

Safety is a requirement of paramount importance for nuclear power. One of the KNGR development policies is to increase the level of safety dramatically. Safety and economics in nuclear power plants are not counteracting each other but really acting in the same direction, since an enhancement of safety will also yield an improved protection of the owner's investment. Therefore, safety has been given top priority in developing the new design. To implement this policy, the plant will in addition to being designed in accordance with the established licensing design basis to meet the licensing rules also be designed with an additional safety margin in order to improve the protection of the investment, as well as the protection of the public health.

The safety objectives for the KNGR can qualitatively be summarized as follows. There shall be:

- No need for short-term (about 24 hours) off-site countermeasures;
- No need for population evacuation beyond site boundary; and
- No long-term relocation of population and restriction of the land use in the vicinity of the plant.

In order to implement this qualitative statement, qualitative safety goals for the designs have been established in a probabilistic approach:

- The total core damage frequency should not exceed 10E-5 per year, considering both internal and external initiating events. In addition, the frequency of core damage with reactor coolant pressure remaining high should not exceed 10E-6 per year.
- The whole body dose for a person at the site boundary should not exceed 0.01 Sv (1 rem) during 24 hours after initiation of core damage, even in the event of containment failure. The probability frequency of exceeding such a limit should be less than 10E-6 per year.
- The frequency of an accident in which the release of long-lived radioisotopes such as Cs-137 would exceed the amount that would limit the land use shall be less than 10E-6 per year.

To achieve the above quantitative goals, the defence-in-depth concept remains as a fundamental principle of safety, requiring a balance between accident prevention and mitigation. With respect to accident prevention, the increased design margins and system simplifications represent important aspects; accident mitigation considerations call for incorporation of design features to cope with severe accidents as well as design basis accidents.

In addition to the public safety, a concept of investment protection will be implemented in the KNGR design. An example of a design requirement that aims at investment protection is the stipulation that a small break loss-of-coolant-accident (LOCA) with a break size smaller than 150 mm in diameter should allow the continued use of the reactor with its fuel inventory after repair of the ruptured pipe (and/or other damages in the reactor coolant system).

Another important design philosophy for safety is the increased design margins. From a utility's point of view, the increased design margin has often been interpreted as an economic loss since a margin means under-utilization of a given capability. However, one of the lessons learnt through more than 30 years of operation of nuclear power plants is that an enhanced margin could benefit the operability and availability of the nuclear power plants. The margin can for example alleviate transients, thereby avoiding

unexpected trips, and be used for later system modification or adaptation of a new fuel design. A few examples of the design requirements following this philosophy are the requested core thermal margin of 10~15%, sufficient system capacity for operator recovery action time of more than 30 minutes, and station blackout coping time of 8 hours.

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

The safety systems consist of the safety injection system, safety depressurization system, in-containment refuelling water storage system, shutdown cooling system, emergency feedwater supply system, and containment spray system. A schematic diagram of arrangements and locations of safety systems is shown in Figure 4.7-1.

Safety injection system (SIS)

The main design concept of the Safety injection system (SIS) is simplification and diversity to achieve higher reliability and better performance. The safety injection lines are mechanically 4 trains and electrically 2 divisions without the tie branch between the injection lines for simplicity and independence. To satisfy the LOCA performance requirements, for breaks larger than the size of an direct vessel injection line (i.e., 216 mm), each train provides 50% of the minimum injection flow rate and, for breaks equal to or smaller than the size of an injection line, each train has 100% of the required capacity. The common header currently installed in the SIS trains is eliminated and, finally, functions for safety injection and shutdown cooling are separated.

Through the In-containment refuelling water storage tank (IRWST) system, the current operation modes of high pressure, low pressure, and re-circulation can be merged into only one operation mode (i.e., safety injection). Accordingly, low pressure pumps are eliminated from the SIS and water source for the safety injection takes from IRWST only. The emergency cooling water is designed to be injected directly into the reactor vessel so that the possibility of spill of the injected flow through the broken cold leg is eliminated. For this purpose four DVI lines are connected with the reactor vessel. The DVI lines will be installed between the hot and cold legs and will be located above the locations of the hot and cold legs.

In-containment refuelling water storage system

The refuelling water storage tank (IRWST) is located inside the containment and the arrangement is made in such a way that the injected emergency cooling water can return to the IRWST. It is shaped as an annular cylinder surrounding the reactor vessel, holdup volume tank (HVT), and reactor cavity. The susceptibility of the current refuelling water storage tank to external hazard is lowered by locating it inside the containment. The IRWST provides the functions of storage of refuelling water, a single source of water for the safety injection, shutdown cooling, and containment spray pumps.

The IRWST is also used as a heat sink to condensing steam discharged from the pressurizer for rapid depressurization if necessary to prevent high pressure core melt or to enable feed and bleed operation. Moreover, it provides the function of coolant supply to the cavity flooding system in case of severe accidents to protect against the molten core. The volume of the IRWST is 2536 m^3 (670 000 gal). This capacity is sufficient for flooding the refuelling cavity during normal operations, assuming the initial RCS level is at the center line of the hot leg. It also covers the capacity (i.e., 833 m^3) to flood the HVT and the reactor cavity to mitigate severe accident impact.

Shutdown cooling system (SCS)

To improve shutdown cooling capability and system reliability, and to remove any possibility of intersystem LOCA, the following improvements are under consideration for the design of the SCS:

- increase of the design pressure to 6.2 MPa for higher system reliability;
- reinforcement of decay heat removal function for the conditions of reactor emergency;
- adoption of the partial 4 train concept by introducing exchangeable shutdown cooling and containment spray pumps;
- installation of the independent heat exchanger; and
- reliability increase by replacing the thermal relief valve with a check valve.

Emergency feedwater supply system

The Emergency feedwater supply system (EFWS) is designed to supply feedwater to the SGs for removal of heat from the RCS for events in which the main or startup feedwater systems are unavailable. In addition, the EFWS refills the SGs following a LOCA to minimize leakage through pre-existing tube leaks. The emergency feedwater system is a 2 divisions and 4 trains system like the SIS. The reliability of the EFWS has been increased by use of two 100% motor-driven pumps, two 100% turbine-driven pumps and two independent safety-related emergency feedwater storage tanks as a water source instead of using the condensate storage tank. The final goal of the design is to achieve the capability of the emergency water supply in 60 minutes after actuation signal and 30 minutes of redundant time for operator action.

Other system improvements

Other system improvements include the containment spray pump and RCP seal water injection pump. The containment spray pump and shutdown cooling pump are designed to be exchangeable to back up each other. The RCP seal is maintained by water injection from the dedicated injection pump to ensure the integrity of the seal.

KNGR is an evolutionary plant which relies on active systems for its safety. However, some passive systems are under research to be incorporated in the design. The three main passive systems under consideration and development are the following:

- The first is a fluidic device which is located at the discharge of the safety injection tank (SIT). It is a passive system to inject the borated water into the RCS in a passively regulating way with a capability of reducing the discharge flow to 10% of the maximum flow. This system is expected to help extend the allowable start-up time of the emergency diesel generators and enhance the performance against loss of coolant accidents.
- The second is the Passive secondary condensing system (PSCS) which is to secure the heat removal through the steam generators in case of a loss of feedwater event. The PSCS is located outside the containment well above the steam generators. The PSCS takes inlet flow into the isolation condenser submerged in the condenser tank from the steam line and discharges outlet flow into the feedwater line. This connection makes it possible to have passive core cooling without any active component running. Moreover, the condenser tank can be used as a backup of the EFWS.
- The last is the adoption of fusible plugs in the reactor cavity. The reactor cavity and the IRWST are directly connected with each other. The IRWST side is for coolant supply and the reactor cavity side is normally closed by end plugs which are made of fusible metal. In an accident situation, if the ambient temperature of the reactor cavity has increased to the melting temperature of the fusible metal, the plugs would start melting and supply cooling water from the IRWST to the reactor cavity.

4.7.6.3 Severe accidents (Beyond design basis accidents)

The most advanced feature for safety from the current nuclear power plant design may be the inclusion of severe accidents mitigation in the design. All ALWRs currently under development have design features addressing severe accident issues in one way or the other. In the Korean next generation reactor, severe accidents are addressed as follows:

- For phenomena likely to cause early containment failure, for instance, within 24 hours after accidents, mitigation systems shall be provided or design should address the phenomena although the probability for such accidents is low.
- For phenomena which potentially lead to late containment failure if not properly mitigated, the mitigation system or design measures should be considered in conjunction with the probabilistic safety goal and cost for incorporating such features to address the phenomena.

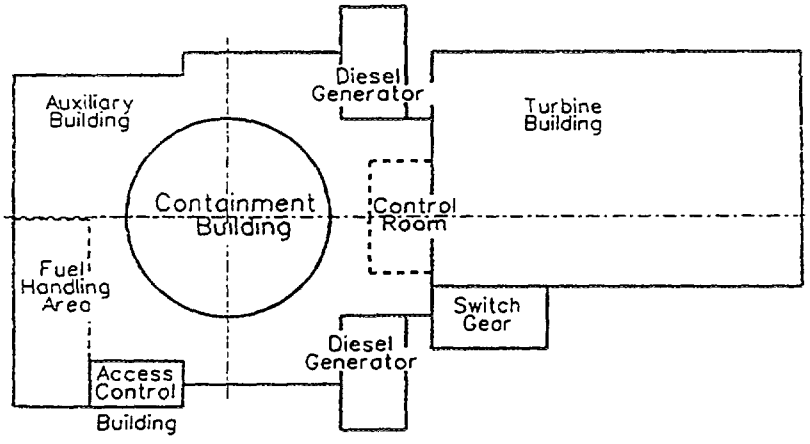


Figure 4.7-7 KNGR - Plant general arrangement

This approach is to enhance the effectiveness of investment on safety by avoiding undue over-investment on highly improbable accidents. Also, a realistic assessment is recommended for severe accident analyses. More specific design features for mitigation of severe accidents are as follows:

- Enhanced capabilities against severe accidents
  - secondary cooling for transients by Passive secondary condensing system to back up the EFWS
  - SBO coping by AAC facility
- Robust double containment with large volume
- Systems dedicated for severe accident mitigation
  - reactor cavity design improvements
  - reactor cavity flooding system with active valves and passive fusible plugs
  - hydrogen igniter (e.g., glow plug and catalytic type).

4.7.7 Plant layout

4.7.7.1 Buildings and structures, including plot plan

The general arrangement of the KNGR has been developed based on the twin-unit concept and slide-along arrangement with common facilities such as the radwaste building. The auxiliary building which accommodates the safety systems and components surrounds the containment building. The auxiliary and containment buildings will be built on a common basemat. The common basemat will improve the resistance against seismic events and reduce the number of walls between buildings so that rebar and formwork cost can be lowered.

The KNGR plant consists of the nuclear island and turbine island:

- Nuclear island: containment building, auxiliary building, diesel generator building, access control building and radwaste building.
- Turbine island: turbine building and annex building.

The general arrangement of the buildings is schematically depicted in Figures 4.7-7, 4.7-8 and 4.7-9. The layout is highly influenced by safety considerations, in particular, by the physical separation of equipment for the safety systems. The safety injection pumps are located in the auxiliary building in the four quadrants, one pump in each, surrounding the containment.

This arrangement ensures physical separation of the pumps, minimizing the propagation of damage due to fire, sabotage, and internal flooding. The emergency diesel generator buildings are also separated and located at symmetrically opposite sides.

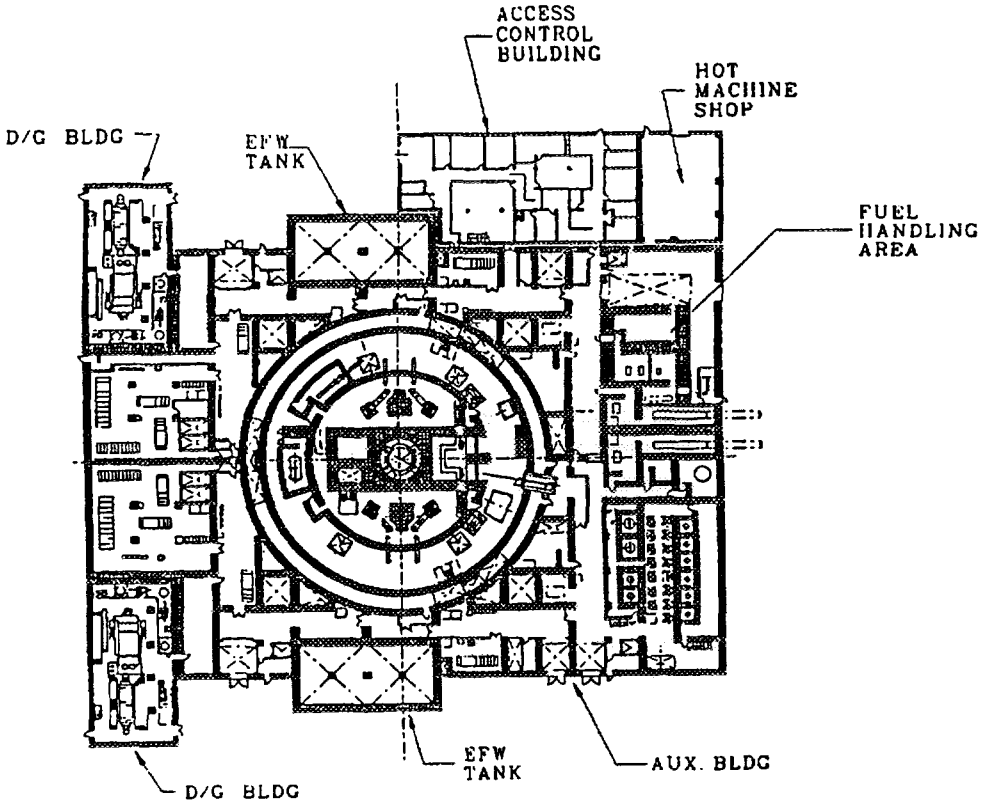


Figure 4.7-8 KNGR - Plant general arrangement plan at EL. 114'-6"

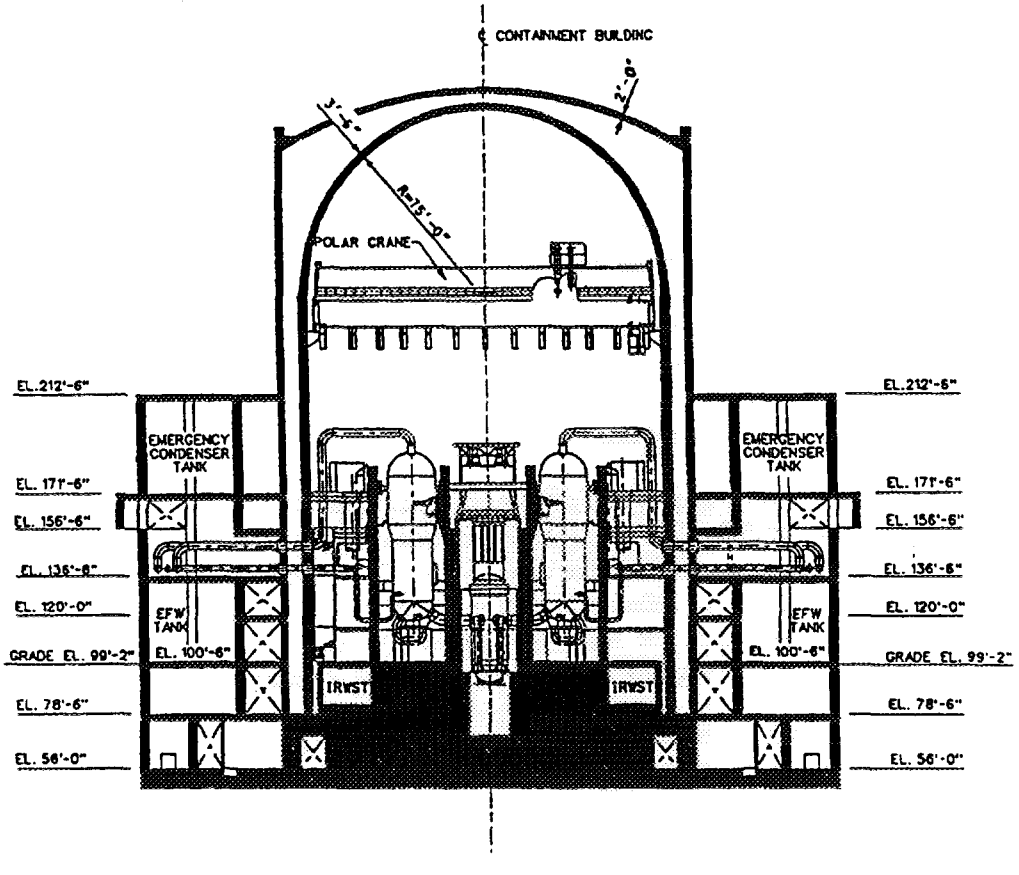


Figure 4.7-9 KNGR - Cross-sectional view of the reactor building (vertical direction)

The building arrangement is also designed for the convenience of maintenance, considering accessibility to and replaceability of equipment. The internal layout of the containment, in particular, is designed to allow the one-piece removal of the steam generator. With proper shielding and arrangement of maintenance space, and careful routing of ventilation air flow, the occupational radiation exposure is expected to be lower than 1 man-Sievert a year.

The design strength of the buildings in the safety category, which are the reactor building with the containment, diesel buildings, and the auxiliary buildings, is sufficient to withstand the effects of earthquakes up to the safe shutdown earthquake (SSE) of 0.3 g

4.7.7.2 Reactor building

The reactor building is the central building of the plant. The KNGR is a pressurized water reactor, and the reactor building essentially coincides with the containment building. Figure 4.7-9 shows a cross-sectional view of the reactor building including a part of the auxiliary building in the vertical direction with the arrangement of major equipment.

4.7.7.3 Containment

The containment building consists of an outer containment in reinforced concrete, an inner containment in post-tensioned concrete with a steel liner, and an internal structure in reinforced concrete. The containment building houses the pressurized water reactor, steam generators, reactor coolant loops, In-containment refuelling water storage tank (IRWST), and portions of the auxiliary systems.

The containment building is designed to provide biological shielding, external missile protection, and to sustain all internal and external loading conditions which may reasonably be expected to occur during the life of the plant. The containment building is on a common basemat which forms a monolithic structure with the auxiliary building.

The interior arrangement of the containment building is designed to meet the requirements for all anticipated conditions of operations and maintenance, including new and spent fuel handling. There are four main floor levels in the containment: the lowest floor level, called the basement, the highest floor elevation, called the operation floor, and two (2) mezzanine floors in between the basement and operating floors are concrete. The two mezzanine floors consist primarily of steel-supported grating.

The equipment hatch is at the operating floor level, and has an inside diameter of 7.8 m (26 feet). This hatch size is selected to accommodate one-piece replacement of a steam generator. A polar bridge crane is supported from the containment wall. The bridge crane has the capability to install and remove the steam generators. The north-south centerline of the reactor vessel is offset from the north-south centerline of the containment by a distance of one foot to allow the polar crane to align with the reactor vessel center. Personnel access to the containment is through two hatches, one located at the operating floor level and the other at the basement floor level, which is also the plant grade elevation.

The outer containment is composed of a reinforced concrete straight cylinder with a shallow, domed roof. It has an inner radius of 25.8 m (86 feet). An annular space, called the annulus, is provided between the inner radius of the outer containment and the outer radius of the inner containment above the basemat. The main function of the annulus is to collect post-LOCA containment atmosphere leakage. This leakage is filtered, recirculated, and released by the Annulus ventilation system. Adequate access is provided for installing, testing, inspecting, and tensioning the tendons of the inner containment structure.

The inner containment is a post-tensioned concrete cylinder with an internal diameter of 45.7 m (150 ft) and a hemispherical top dome. There is no structural connection between the free standing portion of the inner containment and the adjacent structures other than penetrations and their supports. The lateral loads due to seismic and other forces are transferred to the foundation concrete through the structural concrete reinforcing connections. The containment free volume has been increased to meet the EPRI URD require-

ments for hydrogen control, i.e., the volume concentration of hydrogen shall be lower than 13% with 75% of the clad metal melted, and set at 9.1×104 m^3 (3.2×10^6 ft^3).

The EPRI URD requirements for primary containment structure call for a free standing cylindrical steel pressure vessel and a reinforced concrete outer containment. The KNGR inner containment is steel lined, post-tensioned concrete. This arrangement meets the intent of the URD, however, in that it allows for an annular space between inner and outer containments. This annular space permits filtration of leakage from the inner containment, thus ensuring the site dose rates are attainable. The design details of the containment are as follows :

- Inner containment dimensions:
  - Inside diameter : 45.7 m (150 ft)
  - Inside height : 76.7 m (251.5 ft)
  - Wall thickness : 1.2 m (4 ft)
  - Dome thickness : 1.1 m (3.5 ft)
- Outer containment dimensions:
  - Inside diameter : 52.4 m (172 ft)
  - Inside height : 78.9 m (259 ft)
  - Wall thickness : 1.2 m (4 ft)
  - Dome thickness : 0.6 m (2 ft)
- Annulus: 2.1 m (7 ft)

4.7.7.4 Turbine building

The turbine building houses the turbine generator, the condenser systems, the preheater system, the condensate and feedwater systems, and other systems associated with power generation. The turbine building configuration is simplified for constructability, and the maintainability of the systems is improved by centralizing the condensate polishing system, separating the switchgear building and rearranging the equipment hatches. There are four main floor levels referred to as the Basement, the Ground level, the Operating level, and the Deareator level.

The turbine building is classed as non-safety related. It has no major structural interface with other buildings except for a seismic interface with the connecting auxiliary building. It is designed such that under SSE conditions, its failure will not cause failure of safety related structures. The turbine building is located such that the containment building is on the projection of the turbine shaft, on the high pressure turbine side. This allows for optimization of the piping and cable routes to the Nuclear Island. This arrangement also minimizes the risk of damage to safety-related equipment by missiles from the turbine or the generator, in the event of an accident.

The viibration problem which occurs during transient loading is minimized by moving the fresh water tank of the steam generator blowdown system to the auxiliary building.

4.7.7.5 Other buildings

Auxiliary building arrangement

The auxiliary building adjoins the containment building and includes the main control room area, electrical and control area, mechanical areas, which provide control and support functions to the containment building. A major goal in the design development of these areas is to create a safe and efficient environment for the people who work in the plant.

The auxiliary building is on a common basemat which forms a monolithic structure with the containment building. The diesel generator building is built on an isolated mat from other buildings to prevent the propagation of vibration. To assure the safety and reliability, the auxiliary building is designed to enhance physical separation for mitigation of internal flooding, fire propagation as well as security and

sabotage. The auxiliary building shares with the fuel building in a quadrant arrangement. The auxiliary building houses pumps and heat exchangers for safety injection system and safety cooling system. Also, the emergency feedwater tanks and main control room are located in the auxiliary building. For the convenience of operation and maintenance, there is a staging service area in the auxiliary building for installation work in front of the equipment hatch of the containment.

Radwaste building arrangement

The radwaste building houses the liquid waste, gaseous waste, and solid waste systems. The radwaste building also provides structures necessary for the operation of these systems. Radiation shielding is provided where required. The radwaste building is designed to be shared between two units. The radwaste building is classed as non-safety related. In accordance with US NRC Regulatory Guide 1.143, it is designed to provide protection against natural phenomena and to accommodate associated environmental conditions to the extent necssary to retain the spillage of potentially contaminated solids or liquids within the building. It has no major structural interface with other buildings.

Emergency diesel generator buildings

The emergency diesel generator (EDG) buildings are located one on either side of the auxiliary building. These buildings are seismic category I structures which provide protection from fire, missiles, and the environment. The fuel storage tanks are located apart from, and on each side of the auxiliary building. The EDGs are located at ground level. Each is a completely separate entity with dedicated auxiliaries. The buildings are designed for equipment removal through hatches in the roof and walls.

The two EDG buildings each contain an EDG and its auxiliary equipment. Each EDG supports one division. Each building houses the starting air receiver tanks, starting air compressors and after coolers, jacket water and lube oil systems equipment, a fuel oil day tank, and the exhaust silencer. A removable panel is provided on each EDG building outside wall for silencer removal purposes.

The basemat for the EDG buildings is at grade. A 1.8 m (6 in) gap is provided between each EDG building basemat and the auxiliary building basemat. This will prevent vibration of the auxiliary building during EDG operation. Further, the EDGs are arranged as separate entities with dedicated auxiliaries including air supply, exhausts, and cooling systems, so that they are independent of each other in all respects. The EDG buildings are arranged to provide routine maintenance facilities and maintenance access space such that work on one EDG in no way affects the operability of the other EDG. The arrangement of structural facilities is such as to permit the removal and replacement of a complete EDG while the other(s) remain operable.

Access control building arrangement

The access control building is simplified in design only to have a function of access control for security. The design of the building is improved for the radiation control by segregating the accesses to the radioactive and non-radioactive areas.

Hot machine shop

The hot machine shop is located near the auxiliary building to facilitate easy transfer of contaminated equipment and materials.

Switchgear area

The switchgear area is located in the turbine building and all the electrical switchgears are centralized in this area for the convenience of maintenance and efficiency of space allocation.

4.7.8 Technical data

General plant data		
Power plant output, gross*)	1 350	MWe
Power plant output, net*)	1 300	MWe
Reactor thermal output*)	4 000	MW
Power plant efficiency, net		%
Cooling water temperature		°C
Nuclear steam supply system		
Number of coolant loops	2	
Primary circuit volume, including pressuriser*)	448.4	m^3
Steam flow rate at nominal conditions*)	1079	kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	285/6.9	°C/MPa
Feedwater temperature/pressure	232.2/	°C/MPa
Reactor coolant system		
Primary coolant flow rate	20 891	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	291.1	°C
Coolant outlet temperature, at RPV outlet	323.9	°C
Mean temperature rise across core		°C
Reactor core		
Active core height	3.81	m
Equivalent core diameter	3.65	m
Heat transfer surface in the core*)	6 359.9	m^2
Fuel inventory		t U
Average linear heat rate*)	18.14	kW/m
Average fuel power density		kW/kg U
Average core power density (volumetric)*)	98.4	kW/l
Thermal heat flux, Fq*)	2.35	kW/m^2
Enthalpy rise, FH*)	1.55	

*) The exact number could change as the design progresses.

Fuel material	Sintered UO2	
Fuel assembly total length*)	4 127.5	mm
Rod array	square, 16x16	
Number of fuel assemblies	241	
Number of fuel rods/assembly	236	
Number of control rod guide tubes*)	12 or 4 per assembly	
Number of spacers*)	11	
Enrichment (range) of first core*)	1.8/2.9/3.7	Wt%
Enrichment of reload fuel at equilibrium core*)	4.7	Wt%
Operating cycle length (fuel cycle length)*)	18	months
Average discharge burnup of fuel*)	60 000	MWd/t
Cladding tube material	Zr-4	
Cladding tube wall thickness*)	0.635	mm
Outer diameter of fuel rods	9.7	mm
Overall weight of assembly*)	662.7	kg
Active length of fuel rods*)	3 810	mm
Burnable absorber, strategy/material	Al_2O_3-B_4C/UO_2-Gd_2O_3/UO_2-Er_2O_3	
Number of control rods	93	
Absorber rods per control assembly*)	4 or 12	
Absorber material (Full/Part strength)	B4C/Ag-In-Cd/Inconel 625	
Drive mechanism		
Positioning rate		steps/min
Soluble neutron absorber	Boron	
Reactor pressure vessel		
Cylindrical shell inner diameter*)	4 630	mm
Wall thickness of cylindrical shell*)	230	mm
Total height*)	15 280	mm
Base material: cylindrical shell	SA 508, Class 2&3	
RPV head	Inconel 690	
Liner	Stainless steel	
Design pressure/temperature	17.2/343.3	MPa/°C
Transport weight (lower part)		t
RPV head		t

Steam generators

Type	Vertical, U-tube	
Number	2	
Heat transfer surface*)	14 663	m^2
Number of heat exchanger tubes*)	12 580	
Tube dimensions (outer diameter/thickness) *)	19.05/1.07	mm
Maximum outer diameter*)	6 172	mm
Total height*)	22 987	mm
Transport weight	905.4	t
Shell and tube sheet material	SA 533 Grade A/B, Class 1, SA 508, Class 2 or 3	
Tube material	Inconel 690	

Reactor coolant pump

Type	Vertical, single-stage, centrifugal pump	
Number	4	
Design pressure/temperature*)	17.2/343.3	MPa/°C
Design flow rate (at operating conditions) *)	7 028	kg/s
Pump head*)	116	m
Power demand at coupling, cold/hot*)	8 948/6 711	kW
Pump casing material	SA 508, Class 2 or 3	
Pump speed	1 190	rpm

Pressuriser

Total volume	68	m^3
Steam volume: full power/zero power*)	34/	m^3
Design pressure/temperature*)	17.2/371.1	MPa/°C
Heating power of the heater rods*)	2 400	kW
Number of heater rods*)	48	
Inner diameter	2 438	mm
Total height	16 459	mm
Material	SA 533 Grade A/B, Class 1, SA 508, Class 2 or 3	

*) The exact number could change as the design progresses.

Transport weight	164.9	t

Pressuriser relief tank (if any)

Not applicable

Total volume	m^3
Design pressure/temperature	MPa/°C
Inner diameter (vessel)	mm
Total height	mm
Material	
Transport weight	t

Primary containment

Type	Prestressed and reinforced concrete	
Overall form (spherical/cyl.)	Cylindrical	
Dimensions (diameter/height)	45.7/70	m
Free volume	91 180	m^3
Design pressure/temperature (DBEs)	480/	kPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate		vol%/day
Is secondary containment provided?	Double containment	

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage	22.8/345	kV
	rated capacity	3x1450	MVA
Plant transformers,	rated voltage	24/14.49, 4.47	kV
	rated capacity	60/80	MVA
Start-up transformer	rated voltage	345/14.49, 4.47	kV
	rated capacity	55/73.3	MVA
Medium voltage busbars (6 kV or 10 kV)		13.8, 4.15	kV
Number of low voltage busbar systems		2	

Standby diesel generating units: number	2	
rated power		MW
Number of diesel-backed busbar systems	1 per DG unit	
Voltage level of these	4.16	kV ac
Number of DC distributions	4/ 1/ 2	
Voltage level of these	125 (Class 1E)/ 250, 125 (Non-1E)	V dc
Number of battery-backed busbar systems	4/ 1/ 2	
Voltage level of these	125 (Class 1E)/ 250, 125 (Non-1E)	V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)*)	in-line, 6 flow, tandem regenerative reheat TC6F-52	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/ 3 LP	
Turbine speed	1800	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	6.9/285	MPa/°C

Generator

Type	4-pole, 1800	rpm
Rated power	1 658.4	MVA
Active power		MW
Voltage	24	kV
Frequency	60	Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type	Once-through, sea water cooling	
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate*)	37 854	m^3 /s
Cooling water temperature*)	35	°C
Condenser pressure	38.1	mmHg

Condensate pumps

Number	3 × 50%	
Flow rate*)	37 854	kg/s
Pump head		
Temperature*)	35/51.1	°C
Pump speed		rpm

Condensate clean-up system

Full flow/part flow	
Filter type	Deep bed, mixed resin ion exchanger

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/ °C

Feedwater pumps

Number	Turbine driven, 3×50%	
Flow rate*)	2 228	kg/s
Pump head		
Feedwater temperature*)	232.2	°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages	7
Redundancies	3 strings, 3 per string

*) The exact number could change as the design progresses.

4.7.9 Project status and planned schedule

The KNGR development project was launched at the end of 1992, organized in three phases related to the development status, and the third phase ends with completion of a detailed standard design. The total project period is eight years, i.e., by the turn of the century the design shall be ready for start construction of the first unit.

Phase I of the project was scheduled to run for the two-year period from the end of 1992 to the end of 1994, and the major activity was to develop top tier design requirements and concepts for the new design. Phase I was finished according to the plans, and is now followed by Phase II which is a three-year programme, running from 1995 to 1998. The major activities of this phase are to develop a basic design for a licensing review, to ensure the safety of the KNGR and thus, its licensibility. The level of design completion by the end of the Phase II is estimated to be around 20% of the total engineering works needed for construction, and commissioning, of a plant. Phase III, finally, is a four-year programme that will run from 1997 to 2000. The one-year overlap between Phase III and Phase II has been included to expedite the detailed design work; detailed design can be initiated as the basic design is fixed. The major activity during this phase is the development of the detailed standard design to a level that will enable accurate costing evaluation and provide adequate investment assurance. At the end of Phase III, the design completion level is expected to be around 60%.

The major activities in the Phase II programme that are currently under way, can be categorized as a basic study, development of detailed user's requirements, regulatory research, development of an information management system, and design activities. Licensing interactions with the regulatory body will also be an important part as the design work proceeds. In addition, some supporting research and development will be conducted to upgrade the in-house engineering capability and investigate design alternatives. Also, major advanced and passive safety features such as the fluidic device, and the passive secondary condensing system will be tested to verify their functions and designs.

References

Final Reports for Research and Development on Next Generation Reactor (Phase I), Korea Electric Power Corporation, December 1994.

Young Sang Choi and Byong Sup Kim, "Progress in Design, Research & Development and Testing of Safety Systems for the Korean Next Generation Reactor", presented at the IAEA Technical Committee Meeting on the Progress of Water-cooled Reactor Design, Piacenza, Italy, May 1995.

Sung Jae Cho and Dong Wook Jemg, "Research Activities and Design Requirements for the Next Generation Reactor in Korea", presented at the International Workshop on Future LWRs, Tokyo, Japan, July 1995.

4.8 SIZEWELL C, NNC, UK

4.8.1 Introduction

The Sizewell C design is based on the Sizewell B reference plant which entered commercial operation in 1995.

The overall plant design was developed by Nuclear Electric and incorporates a Westinghouse design 4 loop PWR producing 3581 MW of heat from the nuclear steam supply system and 1388 MW (gross) of electricity from two turbine generators.

The Sizewell B design was developed from the Standardised Nuclear Unit Power Plant System (SNUPPS) design, of which there are two plants in operation - Wolf Creek and Callaway in the United States. The design incorporates significant enhancement over the SNUPPS design and addresses the licensing requirements for advanced nuclear power plants. The design offers improved operational flexibility; introduces the latest technology, particularly in respect of instrumentation and control; and provides improved ease of operation and maintenance. It includes the lessons learned from the Three Mile Island accident.

The changes made to the SNUPPS design were largely made to accommodate specific UK siting and safety requirements and take account of international developments in utility requirements. These design enhancements arise mainly from a UK requirement that no accident against which the plant has been designed should lead to the need for off-site evacuation, together with the stringent limits set on operator doses in the UK. The changes include:

- The reactor protection system incorporates two diverse systems. This gives increased protection, improvement reliability and allows for a greater degree of self-checking.
- An emergency boration system has been incorporated to provide a diverse means of shutting down the reactor and increase the protection against certain faults.
- A second steam driven pump has been provided in the diverse auxiliary feedwater system.
- An emergency charging system has been provided to protect the reactor coolant pump seals in the event of loss of AC power. A reserve ultimate heat sink is provided to give additional diversity and protection against certain faults.
- Improvements have been incorporated into the emergency core cooling system, including increased capacity accumulators, four rather than two larger high head injection pumps and improved automatic controls.
- In order to give enhanced protection against such hazards as fire, flooding and missile impacts, four physically segregated trains of redundant protection and safeguards equipment have been provided.
- Four-fold redundancy has been provided in the auxiliary diesel generators which can supply power to those systems necessary to achieve or maintain safe shutdown of the reactor.
- The advanced control room design allows for easier operation of the plant and greater human reliability of operation.
- The containment system includes a secondary containment to reduce off-site doses in the event of a serious accident and the primary containment has been enlarged. A wet cavity design has been adopted to give protection against containment melt-through in the event of severe accidents.
- Measures to achieve low radiation doses to operators include: improved primary circuit chemistry; the use of remote inspection and manipulation equipment; provision of permanent platforms and shielding; a reduction in the use of cobalt, nickel and stellite material; introduction of a refuelling pool sparge ring; and provision of more space in the reactor building.

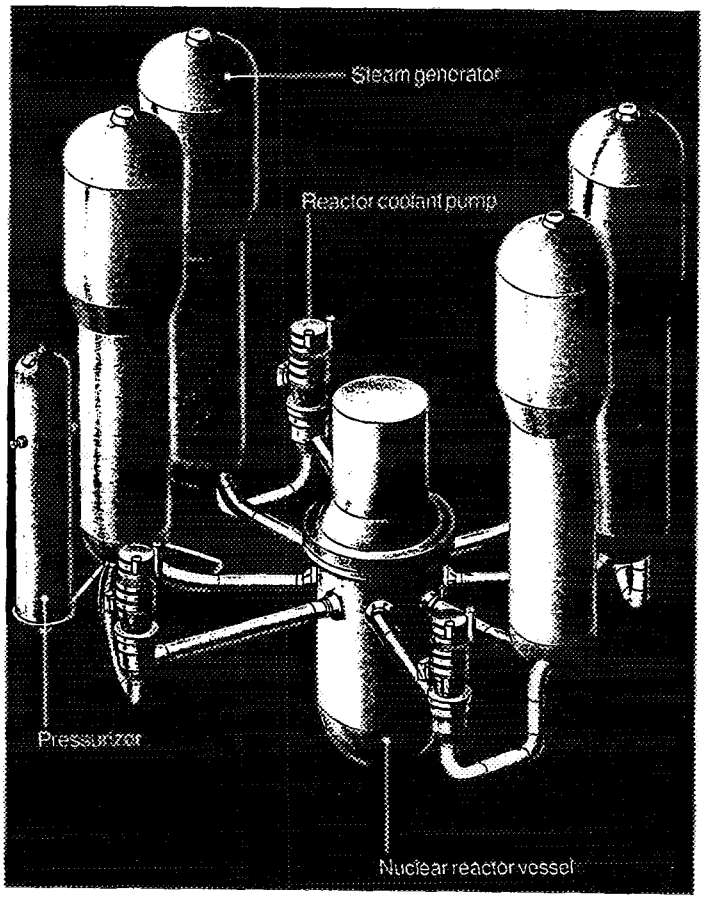


Figure 4.8-1 Sizewell C - Primary system configuration

4.8.2 Description of the nuclear systems

4.8.2.1 Primary circuit and its main characteristics

The primary loop configuration (Figures 4.8-1 and -2) is the same as that of existing designs and is thus well proven. The reactor coolant system comprises four loops, each containing a reactor coolant pump, a steam generator and the interconnecting pipework.

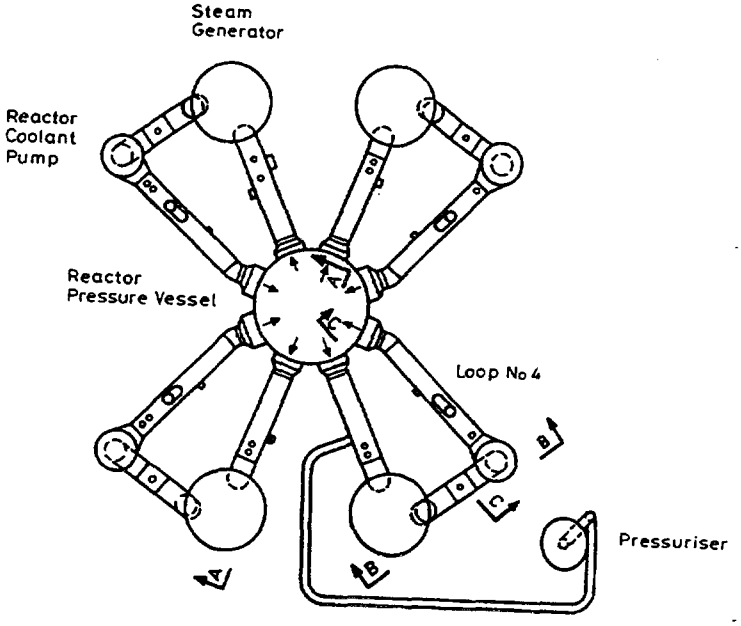


Figure 4.8-2 Sizewell C - Reactor coolant system main loop pipework - plan

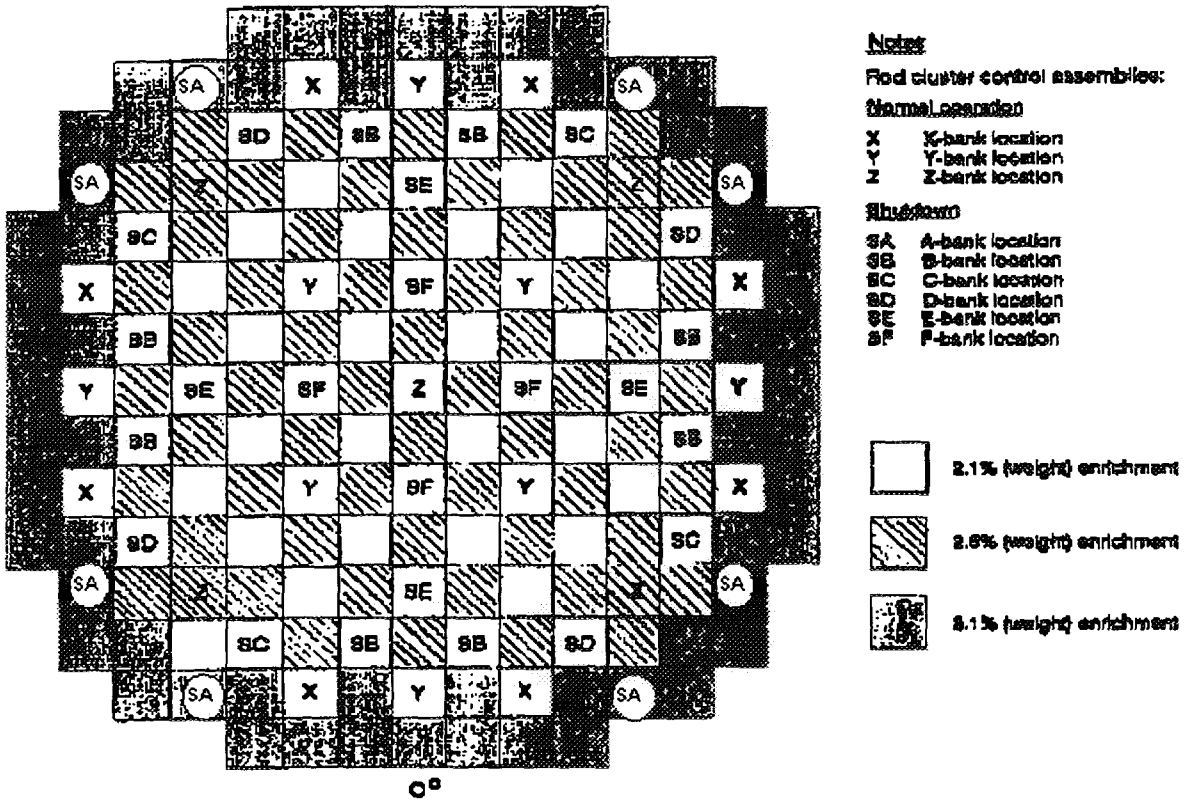


Figure 4.8-3 Sizewell C - Core arrangement during cycle 1 and rod cluster assembly locations

The reactor vessel, pressurizer, and the steam generator shells are constructed of carbon steel of controlled composition and manufacture. The surfaces in contact with reactor coolant are clad with stainless steel to provide corrosion protection. In the case of the reactor pressure vessel this also provides protection against thermal shock in the event of safety injection following large loss of coolant accident (LOCA).

The reactor coolant pump bowls and interconnecting pipes are constructed of stainless steel. The pressure-retaining parts are designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Each component vessel is subjected to a pressure test before installation. After installation, before commissioning, and at prescribed intervals during operational life, the whole pressure boundary is subjected to pressure tests and inspection of welds.

The reactor coolant is normally at approximately 15.5 MPa pressure and 293°C at the inlet to the reactor core. The temperature rise through the core is approximately 33°C. The flow rate through each coolant loop is about 4600 kg/s.

4.8.2.2 Reactor core and fuel design

The nuclear fuel is contained in fuel rods 3.85 m long and 9.5 mm wide in diameter. The fuel rods contain pellets of slightly enriched uranium dioxide in cladding of cold-worked Zircaloy-4 tubing. The core has 193 assemblies. In each assembly, the 264 fuel rods are mechanically located in a 17×17 square array. A cross section of the reactor core is shown in Figure 4.8-3.

Reactivity control is accomplished by changing the boron concentration in the primary coolant and by moving control assemblies. Slow reactivity changes caused by changes of xenon concentration and burnup are compensated by changes of the boron concentration, while fast

reactivity changes for adjustment to the power level are compensated by control rod insertion or withdrawal.

The control assemblies are used for core reactivity contol during normal operation and for reactor shutdown in the event of a reactor trip. There are 53 control rod assemblies, each consisting of 24 absorber rods; the absorber material is silver-indium-cadmium alloy.

Burnable poison assemblies provide a means of temporary reactivity control which is used to compensate for the excess reactivity of fresh fuel. The burnable poison is gadolinium.

4.8.2.3 Fuel handling and transfer systems

The fuel handling and storage system includes the receipt, storage and preparation of new fuel; the preparation and partial dismantling of the reactor prior to refuelling; the refuelling operation; reassembly of the reactor and its preparation for operation after refuelling; the storage of irradiated fuel; and its preparation for transportation off-site in shielded casks. At all times irradiated fuel is handled under borated water.

The refuelling operation begins with the shutdown and cooldown of the plant. Electrical cables are then disconnected from the reactor vessel head. The nuts and studs securing the head are released and removed by a special semi-automatic multi-stud tensioner. The head with its attachments is removed using the containment polar crane and placed on a storage stand. The refuelling cavity is flooded as the reactor vessel head is raised. The CRDMs are decoupled from the rod cluster control assemblies and the upper internals structure is then removed from the reactor vessel and placed on a storage stand in the refuelling cavity. The control rod clusters are left behind in the fuel assemblies. Following removal of the upper internals, the core is exposed and ready for the fuel handling operations in which a refuelling machine lifts the fuel assemblies into and out of the core and into the refuelling cavity. The fuel transfer system rotates the fuel assemblies from the vertical to the horizontal and moves them through a transfer tube in the containment wall, and then returns them to the vertical. A fuel handling machine in the fuel storage pond then lifts the fuel assemblies into the storage racks.

Upon completion of the fuel handling operations, the upper internals structure is installed, control rod drives recoupled, the refuelling cavity is drained and the reactor vessel head with its attachments is replaced and reconnected. This is followed by startup tests, plant heatup and power raising.

4.8.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel is constructed of ring forgings with single-piece dome forgings for the upper and lower heads. The cylindrical section comprises a lower plain ring forging surmounted by a thicker ring containing four inlet and four outlet nozzles. The nozzles are arranged symmetrically around the vessel and connect the vessel to the steam generators and the reactor coolant pumps via the primary coolant piping.

The pressure vessel shell is designed so that changes of section, structural discontinuities and stress concentrations are outside the region subject to high neutron irradiation.

The ring forging design ensures the elimination of the more highly stressed axial welds, and the size of forgings has been maximised to minimise the number of circumferential welds. These welds are located away from the region of highest neutron irradiation and designed to optimise inspection and to minimise the amount of deposited weld metal. The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear in-core instrumentation.

The reactor pressure vessel is internally clad with low carbon stainless steel. The inside surface is ground after cladding to a finish as required for ultrasonic in-service inspection which is performed from inside the vessel.

Reactor internals

The internal structures comprise a lower assembly which supports the core, and an upper assembly which locates the top of the core and provides guidance to the control rods. The two main features of the lower internal assembly are the cylindrical core barrel and a massive lower core support forging which is welded at the bottom of the barrel. The upper internal assembly consists of an upper core support and an upper core plate which is suspended from the upper core support by 50 columns.

The core barrel flange rests on a ledge machined from the flange of the reactor pressure vessel. The fuel assemblies are placed directly on a flat perforated plate, machined from a forging of stainless steel and welded to the core barrel. The fuel assemblies are centred by pins located in the core plate.

The cooling water flows through the core plate through holes dedicated to each fuel assembly. These holes can be calibrated in such a way that a flat flow profile is achieved.

Steam generators

Each loop contains a vertically-mounted Westinghouse Model F U-tube steam generator. The steam generator consists of three sections, a hemispherical bottom head carrying the primary coolant inlet and outlet nozzles, an evaporator section enclosing the U-tube bundle, and an upper section enclosing the moisture separators. The Inconel 690 heat transfer tubes are hydraulically expanded into the forged tubesheet and the ends are seal welded to the tubesheet Inconel cladding. The tubes are supported at intervals by ferritic stainless steel horizontal support plates which have quatrefoil clearance holes to permit flow of the steam/water mixture.

For the outer vessel, axial welds have been eliminated in the shell course, and the head forgings are of single piece construction. The size of all forgings have been maximised to reduce the number of welds to a minimum.

The reliability issues affecting earlier designs of steam generator have been primarily associated with localised tube degradation. The Sizewell C Model F steam generators contain several design and material features to mitigate against such degradation. These features include:

(a) Thermally treated Inconel-690 tubing

(b) U-bend stress relief anneal of the innermost ten rows of tubing

(c) Full depth hydraulic expansion of tubing into the tubesheet

(d) Tube support plates made from Type 405 stainless steel using quatrefoil holes

(e) Reduced sludge deposition due to:

- Flow distribution baffle above tubesheet;
- Feedwater offset at feedring elevation;
- Tube lane blocking devices;
- Increased flow velocities within tube bundle.

(f) Anti-Vibration Bars (AVBs) made from Type 405 stainless steel

(g) Tighter restrictions on tubing, U-bend and AVB dimensions

(h) Improved AVB assembly procedures

(j) Loose parts monitoring and secondary side inspection.

Operating experience from 14 plants with Model F steam generators and nine plants with replacement steam generators with Model F features has shown high reliability. To date there has been no loss of availability associated with tube degradation.

Pressuriser and pressure relief system

The pressuriser is a vertical, cylindrical vessel with hemispherical top and bottom heads, constructed of carbon steel with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant.

During normal operation the pressuriser is partly filled with water and partly with steam at a pressure of about 15.5 MPa (155 bar). Control of the pressure is achieved by the pressuriser heaters and sprays.

If there is an increase in the reactor coolant pressure which is beyond the capacity of the pressuriser spray to control, three pairs of pilot-operated relief valves provide relief. Additionally, protection against overpressure is provided by two spring-loaded safety valves which open by direct fluid pressure action.

Steam discharged through the safety and pilot-operated relief valves is received in a pressuriser relief tank. This tank is located on the floor of the reactor containment. Should the relief tank become overpressurised, it is relieved by bursting discs into the containment.

Reactor coolant pumps

The Westinghouse Model 100 reactor coolant pumps circulate reactor coolant through the reactor vessel to the steam generators for heat removal and return it to the reactor vessel. There is one pump for each coolant loop, located in the cold leg. The pump is a single-stage centrifugal unit driven by a constant speed electric motor at about 1500 rpm (in the UK the electrical supply frequency is 50 Hz).

The pump bowl is of single piece cast stainless steel construction, again to avoid pressure boundary welds. To assure leak-tightness of the shaft a seal assembly is provided with three face-type seals in series to seal against the full internal pressure in the pump.

The motor is a constant-speed, air-cooled, vertical, squirrel-caged induction motor. A fly-wheel situated above the motor ensures that if the pumps trip, they will have sufficient rotational inertia to coast down slowly, thereby providing some core cooling in the initial transient state.

Main coolant lines

The main pipework of the reactor coolant system consists of the hot leg outlet piping from the reactor; the cross leg piping linking the steam generator to the reactor coolant pumps; the cold leg inlet piping; and pipework that links the loops to the pressurizer. It is fabricated from seamless straight legs of pipe and cast bends. The material is austenitic stainless steel.

4.8.2.5 Reactor auxiliary systems

Chemical and volume control system

The CVCS is provided to give control of the chemistry of the reactor coolant and control of the volume (i.e., the inventory) of coolant in the primary circuit.

During normal operation a letdown flow of reactor coolant is continuously discharged from one reactor coolant cold leg to the CVCS where it is chemically processed before being returned to the reactor coolant system (RCS). The purification process removes corrosion products and contaminants in solution and suspension in the reactor coolant and so acts to reduce the radioactivity deposition within the reactor coolant circuit.

The letdown flow is returned from the volume control tank to the RCS by means of charging pumps. Two electrically driven centrifugal charging pumps in parallel are provided to take suction from a volume control tank. The process of letdown and charging are used to control the inventory of coolant in the RCS.

Residual heat removal system

The purpose of the Residual heat removal system (RHRS) is to reduce the temperature of the reactor coolant to cold shutdown temperature at a controlled rate after intermediate shutdown temperatures have been reached using the steam generators, and maintain these conditions during shutdown.

Component cooling water system

The Component cooling water system (CCWS) provides cooling to the following:

- Reactor coolant pump thermal barriers;
- CVCS letdown heat exchangers;
- RHRS heat exchangers;
- Containment fan coolers;
- Fuel storage pond cooling system heat exchangers; and
- Various lubricating oil and other coolers associated with reactor auxiliary system pumps.

The CCWS is arranged in two halves, each equipped with two pumps, a heat exchanger and a surge tank. The system operates in a closed loop with demineralized water, and it is cooled by the essential service water system.

Essential service water system and reserve ultimate heat sink (RUHS)

The essential service water system (ESWS) takes heat from the component cooling water heat exchangers and rejects it to the sea. The system is arranged in two halves, each with two pumps, corresponding to the two halves of the CCWS.

If the ESWS fails, the heat may be rejected instead to the Reserve Ultimate Heat Sink (RUHS). This consists of banks of forced draught water-to-air heat exchangers, with component cooling water on the water side. It is divided into two half systems to correspond with the two halves of the CCWS.

Auxiliary feedwater system

The Auxiliary feedwater system provides feedwater to the steam generators when the main feed pumps are not available. In these circumstances the reactor will not be at power. If main feed fails when the reactor is at power, the reactor is tripped and the Auxiliary feedwater system is designed to start automatically.

Emergency boration system

Although very unlikely, failure of control rods to enter the core when the reactor is tripped is a postulated fault condition. In this event the reactor is shut down by the combination of the negative temperature feedback characteristics of the core, which cause reactivity to reduce if the fuel and reactor coolant temperatures increase, and by the rapid injection of boric acid into the reactor coolant by the Emergency boration system (EBS).

Emergency charging system

The reliability of the seal injection and charging functions of the CVCS has been increased by the provision of an Emergency charging system (ECS). The system, including the pump drive, which is by steam turbine using steam produced in the steam generators or auxiliary boiler, is designed to operate independently of the AC power system and the CCWS. This has been done to increase the diversity and reliability of the systems.

4.8.2.6 Operating characteristics

The Sizewell C control system is designed to the following basis:

1. The plant is capable of following repetitive load changes automatically throughout the range of 15% to 100% of rated power consistent with the cyclic nature of the utility system load demand.
2. The plant is capable of automatically making step changes in load of 10% of rated power and ramp changes of 5% of rated power per minute.
3. The plant is capable of following a load cycle starting at full power, decreasing load to 50% power over a 3 hour period, remaining at 50% power for 2-10 hours and returning to full power over a 3 hour period.
4. The plant has a reserve capability of returning to full power at the design ramp rate at any time during the reference daily load cycle through at least 85% of the equilibrium fuel cycle.
5. The plant is capable of accepting a 50% load reduction from rated power without reactor trip.
6. The plant is capable of accepting complete load rejection from the maximum rated power level without reactor and turbine trip.

4.8.3 Description of turbine generator plant system

4.8.3.1 Turbine generator plant

The twin turbine generators are 3,000 rev/min machines each with a normal full-load gross electrical output of about 700 MW. Each turbine has a double-flow high-pressure cylinder and three identical double flow low pressure cylinders. To avoid excessive wetness, two moisture separators and reheaters are interposed between the high pressure and the low pressure cylinders.

A steam bypass system is provided to enable dumping the full nominal steam flow directly to the main turbine condenser in the event of certain disturbances, in order to avoid pressure surges, and corresponding power peaks, in the reactor.

The generator is of established design. The rotor is cooled with hydrogen and the stator windings are water cooled.

Each low-pressure cylinder has attached to it a condenser in which steam exhausted from the low pressure turbines is condensed and collected as condensate to be returned by condensate extraction pumps and feed pumps to the steam generators. The condensers also serve to accept steam diverted from the turbines through the turbine bypass system following a large loss of turbine generator load, or a reactor trip.

4.8.3.2 Condensate and feedwater systems

The condensate is pumped forward to the deaerator through low pressure heaters and a condensate clean-up system with ion exchange filters by means of three 50% condensate extraction pumps.

The feedwater system consists of the main feed pumps and three stages of high pressure feedwater heaters. There are six electrical motor driven main feed pumps, drawing from the deaerator. Four pumps are required for the full load duty. Drainage from the high pressure heaters is routed to the deaerator. Feedwater flow control is achieved by adjusting feed pump speed and the feedwater flow control valves.

The condensate clean-up system plant consists of chemical purification units located in the condensate train of each turbine between the condensate extraction pumps and the feed heaters.

4.8.3.3 Auxiliary systems

No information provided.

4.8.4 Instrumentation and control systems

4.8.4.1 Design concepts including control room

Design objectives

The design objectives for the control and instrumentation systems are to:

i) permit start-up, operation and safe shutdown of the plant from a centralised point of control with plant variables maintained within limits;
ii) enable reactor safety to be reliability maintained following faults or hazards;
iii) provide a man-machine interface which maximises operator perception and reliability whilst facilitating the discharge of their responsibilities; and which
iv) provide appropriate facilities, including automatic controls and protection, so that the workload on the operator is matched to his capabilities.

Wherever practicable, self-testing redundancy and fault tolerance have been employed to increase the reliability of the control and instrumentation systems. Automatic protection is provided to trip the reactor and to maintain it in a safe state following design basis faults and hazards which require protective actions within 30 minutes.

In order to provide reliable protection, a primary and a secondary protection system have been provided which are independent and diverse. Where equipment, notably sensors, is used for both control and protection, suitable isolation has been provided to eliminate physical interaction between the control and protection functions.

The control and instrumentation equipment has been categorised according to its importance to safety, thereby ensuring that equipment with a specific role has been designed and manufactured to standards commensurate with the effect on safety which failure or faulty operation would have.

The equipment is qualified as appropriate and segregation, redundancy and diversity are provided as necessary to ensure an appropriate level of reliability. The reactor protection system and the control and instrumentation equipment associated with the control of safety category 1 engineered safety features are generally arranged as segregated, four-train systems.

Control room

The main control room is the principal control centre of the plant, from which the majority of the plant items concerned with the operation of the plant can be monitored and controlled during normal operation and following a reactor fault, trip or shutdown.

Manual control and monitoring under normal operating conditions is exercised at the plant control desk and panels by a reactor operator who is able to carry out all necessary minute-to-minute operations for the reactor and turbine generators and their main support systems. Visual display units are situated in the operator's desk. These provide alarms and data presentation from the data processing system. Actions which are needed for reactor safety in a short time-scale of less than about 30 minutes after a reactor trip are automatically initiated.

The operator's desk stands before the main control panel, which provides controls and indications including those for the less frequent situations such as startup, shutdown, and fault conditions,

and for engineered safety features monitoring. Additional visual display units are situated in the main control panel for alarms and data presentation.

The supervisor's desk stands behind the operator's desk. The supervisor performs a supervising function, i.e., ensuring that the correct action is taken during normal and fault conditions.

The auxiliary shutdown room contains the auxiliary shutdown panel which can be used to bring the reactor to a hot shutdown state and maintain it in that condition should the main control room become uninhabitable for any reason. In addition to the displays required for reactor safety, other instrumentation of value to the operator is provided and alternative on-site communications are available. The auxiliary shutdown room is in the auxiliary shutdown building, situated at the opposite side of the reactor building from the main control room.

4.8.4.2 Reactor protection system and other safety systems

The reactor protection system comprises two entirely separate systems: the primary and the secondary protection system. Each of these systems has subsystems which perform the functions of reactor tripping and engineered safety features actuation.

The primary protection system is designed to provide protection against all design basis initiating faults, whilst the secondary protection system provides additional protection against those initiating faults which are frequent, that is, which have an assessed frequency of occurrence of 10^{-3} failures per year or greater.

The primary and secondary protection systems each have four trains of equipment and each train performs both the reactor trip function and the engineered safety features actuation. Both systems have one train of equipment in each of the station separation groups 1, 2, 3 and 4 and are served by separate and diverse essential power supplies and heating, ventilation and air-conditioning for the particular separation group.

Both the primary and secondary systems provide analogue signals for display in the main control room and analogue and status signals to the distributed computer system for information display purposes. The primary system also provides analogue and status signals to the high integrity control system.

The primary protection system uses microprocessor based technology to provide means of tripping the reactor for all faults within the design basis which could lead to unacceptable reactor states. The use of this technology makes it feasible for the primary protection system to cover a wide range of plant faults and to provide improved protection.

The system architecture adopts multiple microprocessor based subsystems with a minimum of interconnections and interfaces between them. This enables the simultaneous execution of programs on separate microprocessors. The software structure is based on a large number of configurable software units, or modules, to perform functions which are frequently and generally used for system level operations. The software incorporates extensive self-checking and diagnostics features which allow the identification of faults within the system. Data is communicated within the system via optical data links with pre-defined message formats and data checking features. This means of data transfer provides enhanced integrity over that which would be achievable using conventional hardwired signals.

The secondary protection system uses technology which has traditionally been used in UK power plants. This technology uses trip amplifiers, pulse to d.c. converters and solid state magnetic logic units known as laddics. Within each train, the system has a guard line cubicle together with parameter measurement equipment and reactor trip switchgear.

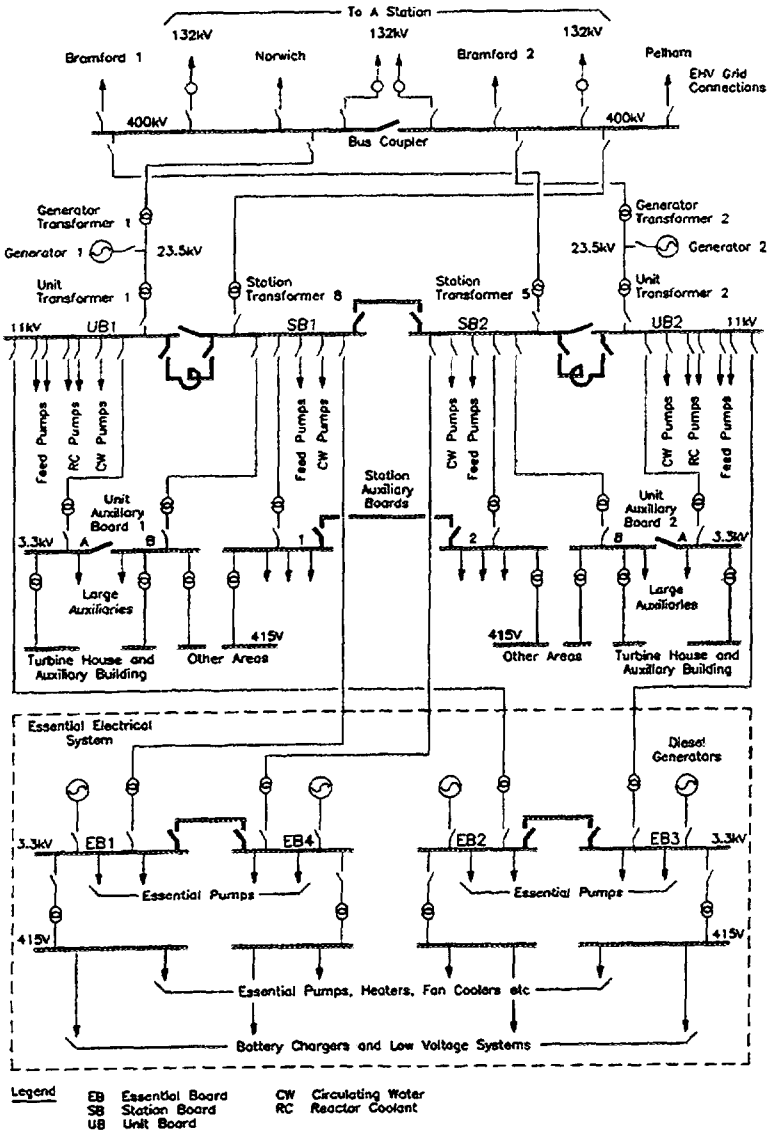


Figure 4.8-4 Sizewell C - Main and essential electrical system diagram

The use of trip amplifiers and magnetic logic units (laddics) enables the secondary trip to provide additional reactor protection which uses a diverse technology from that used in the primary system. All components such as sensors, pressure tappings, sensing lines etc., are dedicated to the secondary system and are entirely separate from the primary protection system.

4.8.5 Electrical systems

A single line diagram for the electrical systems is provided in Figure 4.8-4.

4.8.5.1 Operational power supply systems

High and medium voltage systems

The main generators produce electrical energy at 23.5 kV. This is transmitted from each generator through phase isolated bus-bars to an associated main generator transformer which raises the voltage to 400 kV. Each of the main generators can also supply power to the plant internal distribution system via a unit transformer which reduces the voltage to 11 kV. The 400 kV substation also provides a source of power from the grid to the plant. This power is imported directly via two plant transformers feeding the internal distribution system at 11 kV.

The main AC distribution voltages are 11 kV, 3.3 kV and 415 kV. There are two pairs of 11 kV distribution boards. One of each pair (the "unit" board) is supplied either from a main generator via a unit transformer or from the grid via a generator transformer and a unit transformer; the other (the "plant" board) is supplied from the grid only by a plant transformer. Means are provided for interconnecting the boards in case of failure of any one of the sources of supply. Each of the 11 kV unit boards supplies two main feedwater pumps, two reactor coolant pumps, one main cooling water pump, a non-essential 3.3 kV distribution board and one essential electrical system 3.3 kV distribution board. Each of the 11 kV plant boards supplies one main feedwater pump, one main cooling water pump, two non-essential 3.3 kV distribution boards, and one essential electrical system distribution 3.3 kV board. The non-essential 3.3 kV distribution boards supply a number of auxiliary cooling water, condensate, and other auxiliary pumps and a series of 415 V distribution boards throughout the plant.

Low-voltage systems

Several low voltage systems are required, some AC and some DC. Those important for reactor safety are part of the essential electrical system and are backed by batteries

Auxiliary DC systems serve many other loads which cannot accept a supply interruption without causing damage or loss of station availability, or which for other reasons require a DC supply, but which are not directly required for safety. These loads include: plant computers; most telecommunications; control and switching supplies for general plant operation; and oil pumps to protect the turbines and other major plant equipment.

4.8.5.2 Safety-related systems

Medium voltage systems

The essential electrical system is designed to provide a highly reliable source of power supply for those systems required to ensure the safety of the reactor. The system is based on four 3.3 kV distribution boards, each supplied by a separate route from the grid via an 11 kV plant or unit board, and each being capable of being supplied by one of the four diesel generators. The 3.3 kV boards supply the following pumps: high head safety injection; containment spray; residual heat removal; auxiliary feed; component cooling water; essential service water; and CVCS charging.

Each 3.3 kV board also supplies a number of 415 V boards whose loads include: containment fan coolers; reserve ultimate heat sink fans and pumps; a proportion of the pressurizer heaters; fuel storage pond cooling pump; and battery chargers.

Low-voltage systems

As noted above, there are several low voltage systems, some AC and some DC, and those that are important for reactor safety are are backed by batteries. The principal loads in this category include: reactor protection system; actuation, control and switching supplies for equipment required for reactor safety; essential instrument supplies; emergency lighting for the main control and auxiliary shutdown rooms; and essential communications systems.

Operation without AC power

The essential electrical system is provided with power from diverse sources: the grid and on-site diesel generators. The system is well segregated to avoid total loss due to hazards; it is redundant in equipment, and it supplies safeguards systems which are themselves redundant. Despite this, for the Sizewell C design it is postulated that all AC power supplies could fail for a few hours and sufficient equipment is provided to allow the reactor plant to be kept safe without AC power being available. In particular, two of the auxiliary feed pumps and the two emergency charging pumps are steam driven and their auxiliary equipment designed to operate using DC power from batteries where electrical power is needed. Two battery charging diesels maintain DC supplies in the event of prolonged loss of all AC supplies, each of which is capable of providing all necessary DC supplies.

4.8.6 Safety concept

4.8.6.1 Safety requirements and design philosophy

Safety approach

The nuclear safety strategy which has been adopted for the Sizewell B plant rests on five fundamental principles. These principles are:

i) No person shall receive doses of radiation in excess of the statutory dose limits as a result of normal operation.
ii) The exposure of any person to radiation shall be kept as low as reasonably practicable.
iii) The collective effective dose equivalent to operators and to the general public as a result of the nuclear installation shall be kept as low as is reasonably practicable.
iv) All reasonably practicable steps shall be taken to prevent accidents.
v) All reasonably practicable steps shall be taken to minimise the radiological consequences of any accident.
vi) no accident against which the plant has been designed should lead to the need for off-site evacuation.

These principles are embodied in guidelines used in the design and development of the plant. Incorporated within these guidelines are numerical design targets which ensure that the safety principles can be achieved.

With respect to the risk arising from accidents, the level of acceptability for fatal risks to individual members of the general public has been taken as 10^{-6} per year.

The general approach adopted to ensure that the fundamental principles (items i-v) are met as three main aspects:

a) The design and operating approaches whereby radiation exposure to the operators and the public is maintained as low as is reasonably practicable during normal operation and in any case within prescribed limits.
b) The design, construction and operating approaches whereby faults are prevented from occurring as far as is reasonably practicable and, given the occurrence of a fault, the radiological consequences are rendered as small as is reasonably practicable with the aim of not exceeding certain design targets.
c) The safety analysis approach whereby faults are systematically considered and adequate safety provisions are shown to exist in each case.

Radiological safety during normal operation

The approach to ensure radiological safety during normal operation relies on applying specific control measures which satisfy fundamental principles i-iii.

The basic target which has been adopted relating to the restriction of dose rates to operators is an annual dose equivalent of 10 mSv. In addition, a restriction is imposed on the collective dose to plant staff of 2 man Sv per year per GW(e) installed capacity.

Radiation doses to the public can arise from liquid and gaseous effluent discharges and from direct radiation from plant and buildings. Doses are assessed against a target level which is one-thirtieth of the level recommended by the ICRP.

Avoidance and mitigation of faults

In general, faults are avoided by adopting a well established and controlled approach to the design and construction of the plant. This includes: the categorisation of components and structures according to their safety significance; the use of established codes and standards; consideration of the range of operating conditions in the specification of plant; the choice of appropriate materials;

the qualification of components against specified operating conditions; and the adoption of particularly high standards and requirements for components whose failure is claimed to be incredible.

In addition, to these measures, the mitigation of faults which do occur is facilitated by the systematic and comprehensive identification of potential faults which are then taken account of in the design. This includes the provision of reliable safeguards equipment, incorporating redundancy and diversity as appropriate.

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

Safety systems configuration

The safety of the Sizewell C relies predominantly on the function of active, engineered safety systems in the same way as in currently operating nuclear power plants. In order to attain an enhanced safety level and an enhanced protection against such hazards as fire, flooding and missile impacts, four physically segregated trains of redundant protection and safeguards equipment have been provided.

Emergency core cooling system and safety injection system

An overview on safety injection, residual heat removal and spray systems are provided in Figure 4.8-5.

The emergency core cooling system (ECCS) operates in the event of a LOCA to replace sufficient of the lost coolant and prevent fuel temperature limits being exceeded. As the sequence of events following a LOCA depends on the size and location of the breach in the primary circuit, the ECCS has three separate subsystems which perform different roles: the accumulators and the high head and Low head injection systems.

Following a large LOCA, the core will be uncovered in a matter of seconds and there is therefore a need to deliver a large quantity of water as fast as possible into the circuit. Four accumulators are provided for this purpose.

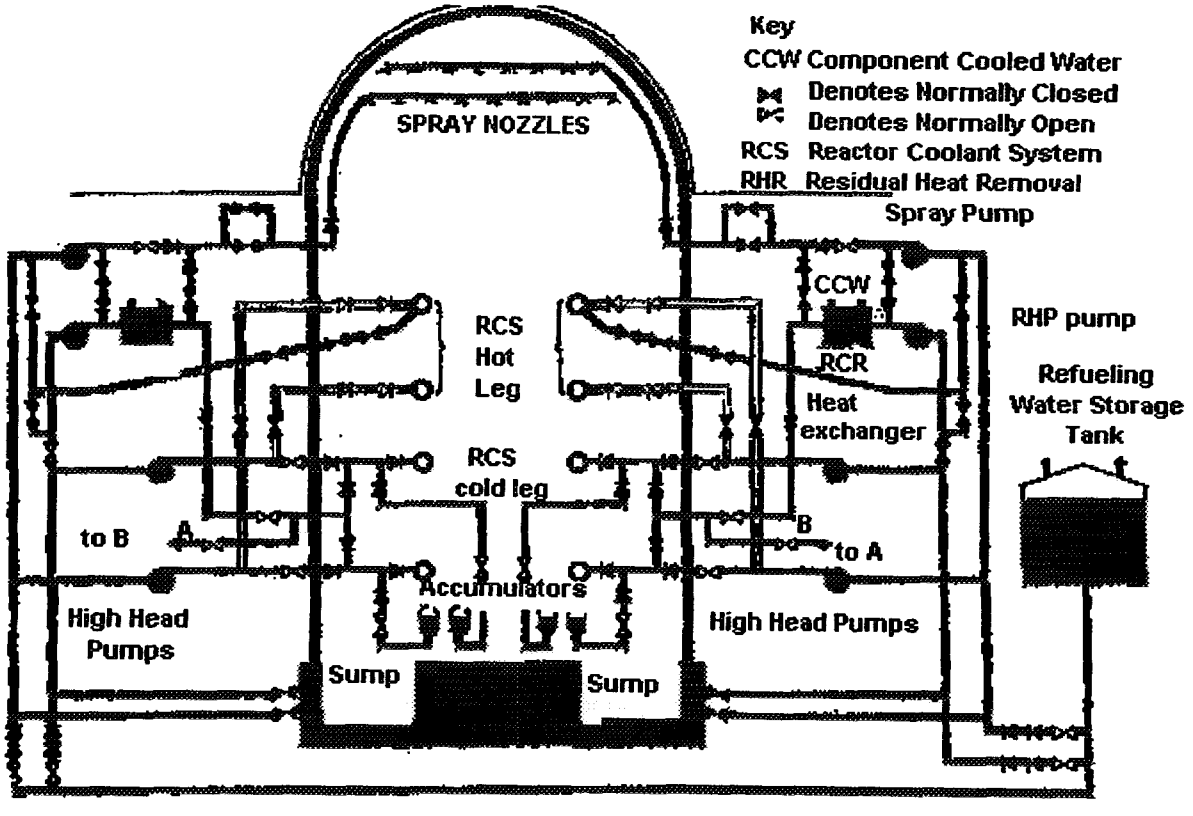


Figure 4.8-5 Sizewell C - Safety injection, residual heat removal and spray systems

As soon as the pressure in the reactor coolant system (RCS) falls below that in the accumulators, non-return valves open and borated water is forced into the primary system by gas pressure. Because no mechanical equipment is required to work apart from the non-return valves, this provision is described as the passive injection system. The other two injection systems both rely on pumps to deliver water to the RCS. The high head safety injection system consists of four pumps connected one to each cold leg. These start to deliver water when the RCS pressure falls below about 12.5 MPa.

In a large LOCA, the accumulators will be rapidly discharged, and further addition of water will be required to complete the process of replenishing the water inventory, and thereafter to replace the losses. This is done partly by the high-head pumps and partly by the low-head safety injection system. The latter system makes use of the two pumps and heat exchangers of the residual heat removal system (RHRS) to deliver water to all four cold legs through a manifold pipe system. It starts to deliver when the pressure falls below about 1.7 MPa.

Both high and low head safety injection systems draw water from the refuelling water storage tank, which is situated outside the containment. When the level in the tank falls below a given limit, both systems are automatically realigned to draw water from the containment sumps where the water spilled from the breach collects. In this way continuous recirculation can be established and can be maintained indefinitely.

Containment spray system

In the event of a LOCA, or a main steam pipe failure within the containment, the containment spray system operates to decrease the containment atmosphere temperature and pressure by heat removal and to remove radioactive iodine from this atmosphere by the scrubbing effect of the spray water. The system contains two pumps that draw water from the refuelling water storage tank and deliver it to two sets of spray nozzles in the upper part of the containment.

Combustible gas control system

To avoid flammable concentrations of hydrogen occurring within the primary containment following a LOCA, the Combustible gas control system provides control by mixing and recombination. Two electrically heated recombiners are provided for this purpose inside the primary containment.

Component cooling water system

The Component cooling water system (CCWS) provides cooling to the following:

- Reactor coolant pump thermal barriers;
- CVCS letdown heat exchangers;
- RHRS heat exchangers;
- Containment fan coolers;
- Fuel storage pond cooling system heat exchangers; and
- Various lubricating oil and other coolers associated with reactor auxiliary system pumps.

The CCWS is arranged in two halves, each equipped with two pumps, a heat exchanger and a surge tank. The system operates in a closed loop with demineralized water, and it is cooled by the essential service water system.

Essential service water system and reserve ultimate heat sink (RUHS)

The essential service water system (ESWS) takes heat from the component cooling water heat exchangers and rejects it to the sea. The system is arranged in two halves, each with two pumps, corresponding to the two halves of the CCWS. If the ESWS fails, the heat may be rejected instead to the reserve ultimate heat sink (RUHS). This consists of banks of forced draught water-to-air heat exchangers, with component cooling water on the water side. It is divided into two half systems to correspond with the two halves of the CCWS.

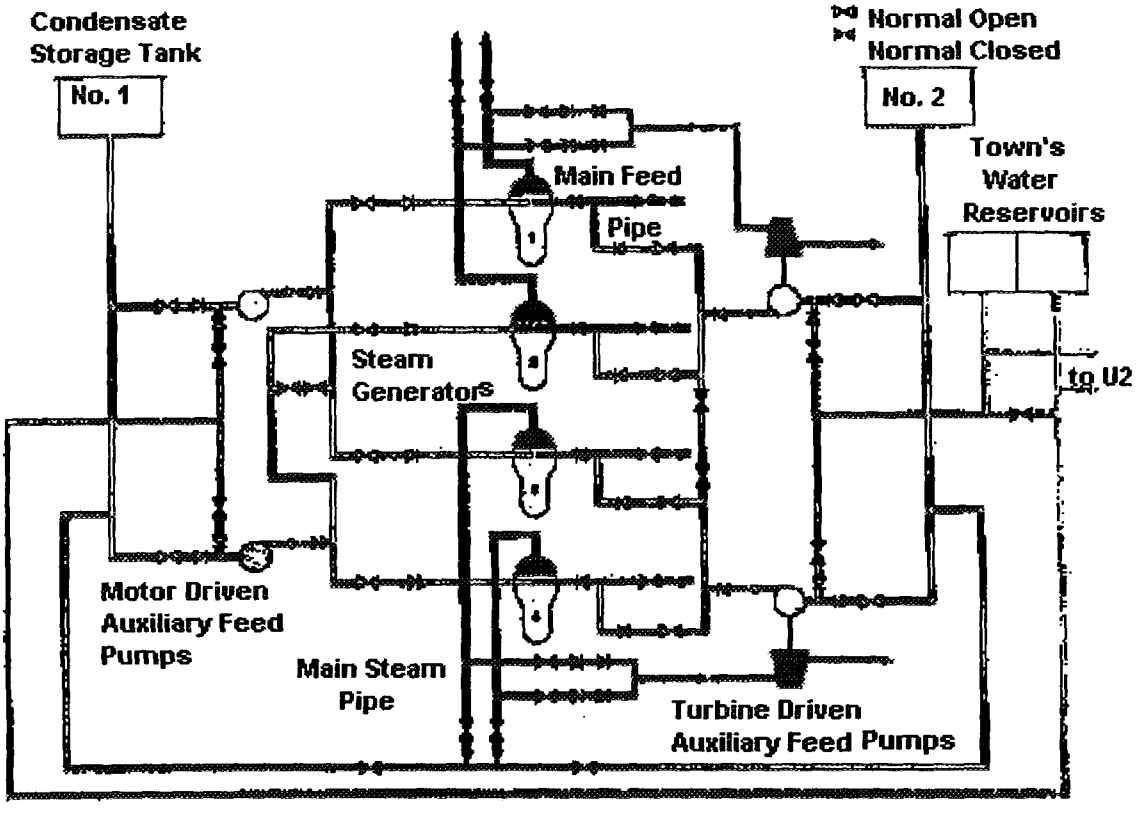


Figure 4.8-6 Sizewell C - Auxiliary feedwater system

Auxiliary feedwater system

The auxiliary feedwater system (Figure 4.8-6) provides feedwater to the steam generators when the main feed pumps are not available. In these circumstances the reactor will not be at power. If main feed fails when the reactor is at power, the reactor is tripped and the Auxiliary feedwater system is designed to start automatically.

The auxiliary feedwater system consists of two independent subsystems of diverse design and having separate water sources. The first subsystem consists of two electric-motor-driven pumps which draw water from one of the two condensate storage tanks or, if this supply is exhausted or unavailable, from the town's water reservoirs. The pumps deliver directly to the four steam generators through nozzles separate from the main feed. The second subsystem is arranged in a similar manner except that the pumps are steam turbine-driven using steam produced in the steam generators.

Emergency boration system

Although very unlikely, failure of control rods to enter the core when the reactor is tripped is a postulated fault condition. In this event the reactor is shut down by the combination of the negative temperature feedback characteristics of the core, which cause reactivity to reduce if the fuel and reactor coolant temperatures increase, and by the rapid injection of boric acid into the reactor coolant by the emergency boration system (EBS) (Figure 4.8-7).

The EBS is situated inside the containment and consists of four stainless steel tanks of concentrated boric acid solution (about 7000 ppm of boron) each of which is connected at one end to a cold leg and at the other end to the pump suction leg of the same primary coolant loop. Normally the tanks are isolated from the RCS by closed power-operated isolating valves in the connecting pipes. If failure of two or more rod cluster control assemblies to insert is detected after a reactor trip demand, the isolating valves are automatically opened and the boric acid is forced into the RCS by the pressure difference between the reactor coolant pump outlet and inlet.

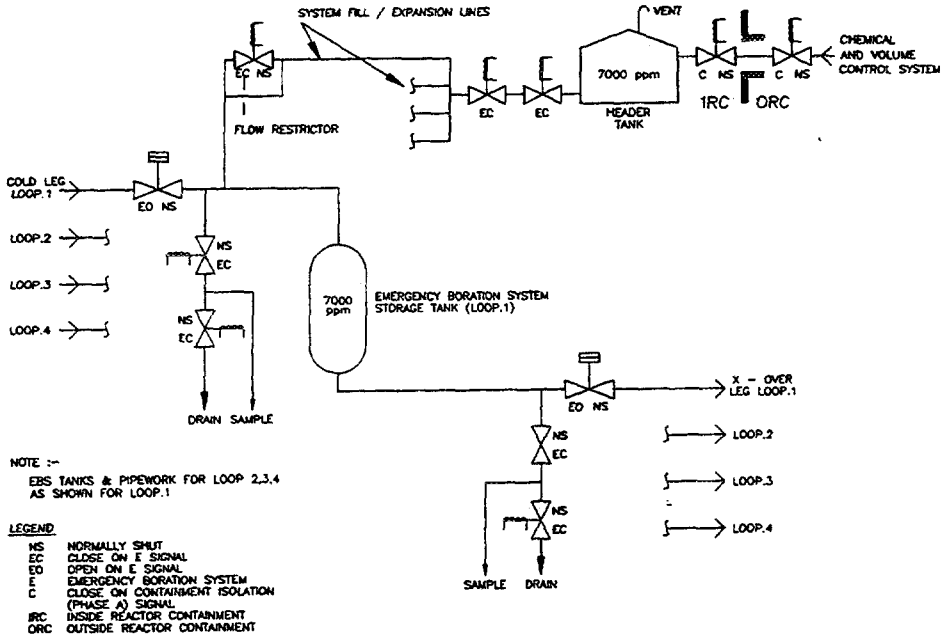


Figure 4.8-7 Sizewell C - Emergency boration system

Emergency charging system

The reliability of the seal injection and charging functions of the CVCS has been increased by the provision of an emergency charging system (ECS) (Figure 4.8-8). The system, including the pump drive, which is by steam turbine using steam produced in the steam generators or auxiliary boiler, is designed to operate independently of the AC power system and the CCWS. This has been done to increase the diversity and reliability of the systems.

Residual heat removal system

The purpose of the residual heat removal system (RHRS) is to reduce the temperature of the reactor coolant to cold shutdown temperature at a controlled rate after intermediate shutdown temperatures have been reached using the steam generators, and maintain these conditions during shutdown.

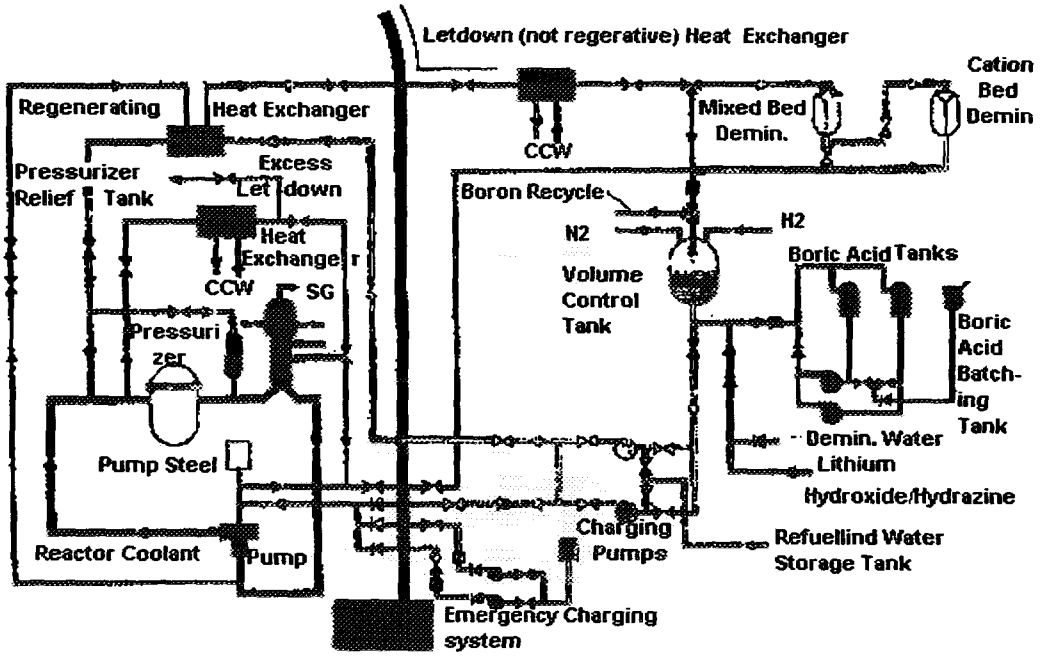


Figure 4.8-8 Sizewell C - Chemical and volume control system and emergency charging system

4.8.6.3 Severe accidents (Beyond design basis accidents)

Faults which exceed the radiological plant safety limit are said to result in an uncontrolled release and are assessed against a target with the aim of ensuring that if met, the underlying target of the level of fatal risk to an individual member of the public is not threatened. The target is a maximum summated frequency of all accidents leading to an uncontrolled release of 10^{-6} per year.

The potential for core damage sequences to progress to the stage whereby they can threaten the containment integrity has been the subject of much research and analysis. Failure of the containment, once isolated, is considered to come about either by over pressurization, missile penetration or by basemat failure as a result of core concrete interaction.

The Sizewell C containment design is able to withstand pressures up to at least twice the design pressure. Analysis has been carried out that shows that the containment will withstand 2.4 times the design pressure before enhanced leakage occurs and 2.6 times the design pressure before gross failure occurs. Failure of reactor building penetrations does not occur until even higher pressures are attained.

Containment overpressurization is prevented by the provision of two diverse and redundant cooling systems. Heat can be removed from the containment atmosphere either by the four safety grade fan coolers or by the reactor building spray system. To provide additional assurance, as part of the severe accident management procedures the operator is instructed to depressurize the system using the SEBIM pilot-operated safety relief valves.

In a severe accident, combustible gases are formed not only by the oxidation of the fuel cladding but also by the erosion of the concrete basemat in those rare sequences where the core is predicted to relocate into the cavity. For these sequences, the hydrogen recombiners would be able to maintain the combustible gas levels below those which would support a deflagration. Under these circumstances, the hydrogen recombiners will act, together with the large reactor building, to ensure that hydrogen concentrations are kept below 10 percent, assuming 100 percent core oxidation. This will minimize the resultant temperature and pressure rise in the reactor building, reducing the level of challenge to the containment to a level which the containment is capable of surviving and negating the possibility of sufficient hydrogen being allowed to build up to support a detonation.

The design includes the provision of a passive means of ensuring that in the event of an accident, water spills over into the reactor cavity so that following vessel failure, the molten core material would be quenched and base mat failure prevented.

4.8.7 Plant layout

4.8.7.1 Buildings and structures, including plot plan

Design requirements

The plant design takes into account the hazards arising from natural phenomena, including extremes of wind, weather conditions and sea level, and seismic disturbances.

The seismic design basis for Sizewell B was an earthquake level which has a return frequency of less than once in 10 000 years (i.e., a probability of occurrence $<1.0E\text{-}4$/year). In the UK, this was established to be 0.25 g, and plant and equipment that have been seismically qualified by analysis have been qualified against this level of peak ground acceleration (PGA). However, components and equipment which have been qualified by test have, in general, been qualified against a PGA of 0.4 g.

Within the nuclear buildings, components and equipment have been designed and located to take into account internal hazards such as fire, flood, pipewhip, jet impingement, missiles, etc.

The plant layout has been optimised from the viewpoint of constructability, operability, maintainability, and inspectability. It has also been rigorously examined and developed to ensure operator doses are as low as reasonably practicable (ALARP). These have all been achieved by interdisciplinary reviews of the model and equipment during the design, layout development, and equipment procurement stages.

Strict targets have been set on occupational radiation exposure to radioactivity for both individual dose and plant collective dose. The target for the latter is 2.4 man Sv (240 man-rem) per year. In practice, the plant collective dose will be less than 1 man Sievert (100 man-rem) per year.

All the design, procurement, construction, and commissioning activities are carried out to the quality requirements of BS 5882 "Specification for a Total Quality Assurance Program for Nuclear Installations" which fully meets the scope and contents 10CFR50 Appendix B and IAEA-Code of Practice 50-C-QA. The entire design has been subject to a comprehensive deterministic and probabilistic safety assessment, and the results from the probabilistic studies have been incorporated into the design.

A comprehensive Level 3 PRA covering all power states has been used to support the safety case. Using fault schedules, data and success criteria more conservative than that recommended by the Electrical Power Research Institute (EPRI), the core melt frequency has been determined to be less than 1.0E-5/year).

Layout

The site layout (Figure 4.8-9) has been established on the basis of economic consideration and the requirement to maintaining a functional relationship between the main power block, ancillary buildings and circulating water structures, while meeting architecture, landscaping, and environmental requirements.

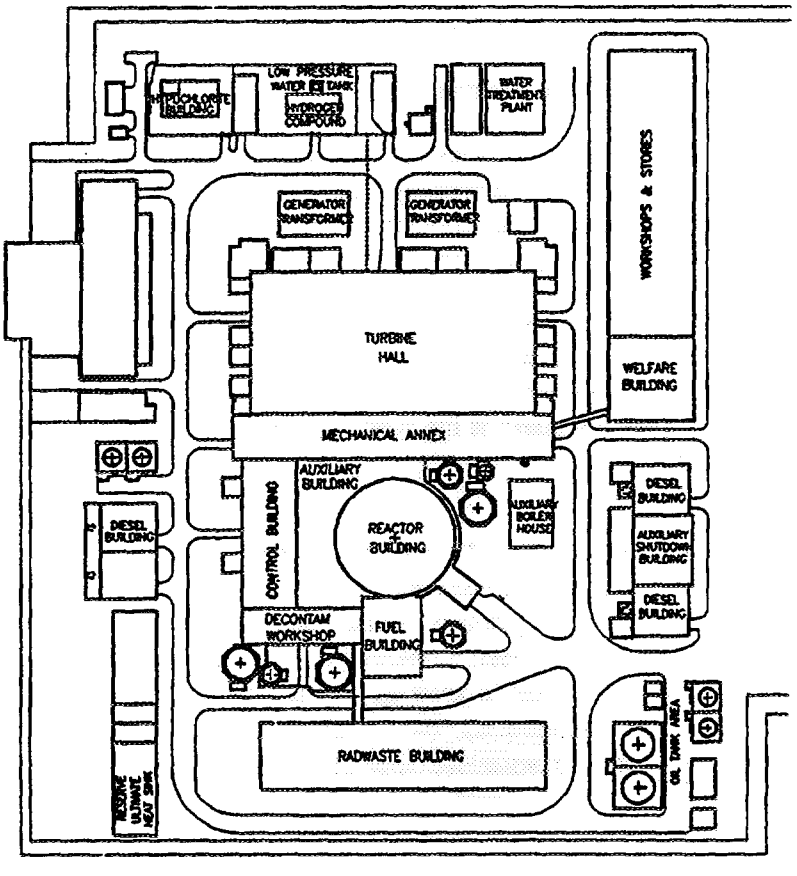


Figure 4.8-9 Sizewell C - Site layout

The main power generating plant and many of its supporting systems are housed in a group of adjoining buildings, which include the following: reactor building (with secondary containment); auxiliary building; control building; fuel building; turbine house; mechanical annex; and decontamination shop.

The area around this main complex, as nominally defined by the site inner ring road, is kept clear of auxiliary buildings except for the auxiliary boiler house, tanks, and transformers.

The two generator transformers are situated outside the south wall of the turbine house and in line with the turbine generator sets they serve. A plant transformer, two unit auxiliary transformers, a plant auxiliary transformer, and an inductor are situated at both the east and west sides of the turbine house, and two essential auxiliary transformers are situated alongside the control building.

Storage tanks situated within the ring road around the main complex are:

- A demineralized water tank, one of the condensate tanks and the emergency charging system tank, all to the west of the reactor building
- The reactor make-up water tank, the refuelling water tank, and the other condensate tank, all to the north of the reactor and auxiliary buildings
- The flask preparation bay water tank, situated to the west of the Fuel Building

North of the main complex is the radioactive waste process and storage building and nitrogen storage compound. To the south of the radioactive waste and process building a service tunnel and bridge provides access to this building from the auxiliary building and decontamination workshop. The fuel oil tanks for the essential diesels (associated with the auxiliary shutdown building) and for auxiliary boiler house are located to the west of the radwaste building on the opposite side of the ring road.

On the east side of the ring road around the main plant complex there are, from north to south, the RUHS, two essential diesel buildings and attendant fuel oil tanks, circulating water surge chamber, and the circulating water pumphouse. Circulating water tunnels run offshore from the circulating water pumphouse and surge chamber.

South of the ring road from east to west are the carbon dioxide store, gas bottle store, the hypochlorite generation building, hypochlorite/hydrogen generation plant switchroom, hydrogen generation plant, hydrogen cylinder storage compound, bulk chemical stores, water treatment plant, garages, two reservoirs, and the fire-fighting pumphouse.

West of the inner road lies the welfare and workshop building, the administration building and the gatehouse. To the north of the welfare building are the auxiliary shutdown and a further two essential diesel buildings, to the west of which lies the 400 kV substation. Just outside the site security fence but within the site boundary is the plant car park.

4.8.7.2 Reactor building

The reactor building (Figure 4.8-10), which comprises the primary containment and a separate secondary enclosure structure, houses the reactor coolant system and some of the reactor auxiliary and safety equipment. The primary containment is a prestressed concrete cylindrical structure with a hemispherical prestressed concrete dome roof. The containment base is of reinforced concrete containing a keyhole shaped slot to accommodate the RPV and its instrument guide tubes at low level. For leaktightness, the concrete structure has a 6 mm thick steel liner attached to its internal surfaces. Penetrations are provided for pipes and cables to enter the containment, and for personnel and equipment access. Each of the two-man access penetrations incorporate two interlocked doors with an air lock between them. The equipment hatch provided for plant construction, repair, and replacement purposes has been sized to permit steam generator replacement. All these penetrations are firmly anchored and sealed into the containment wall.

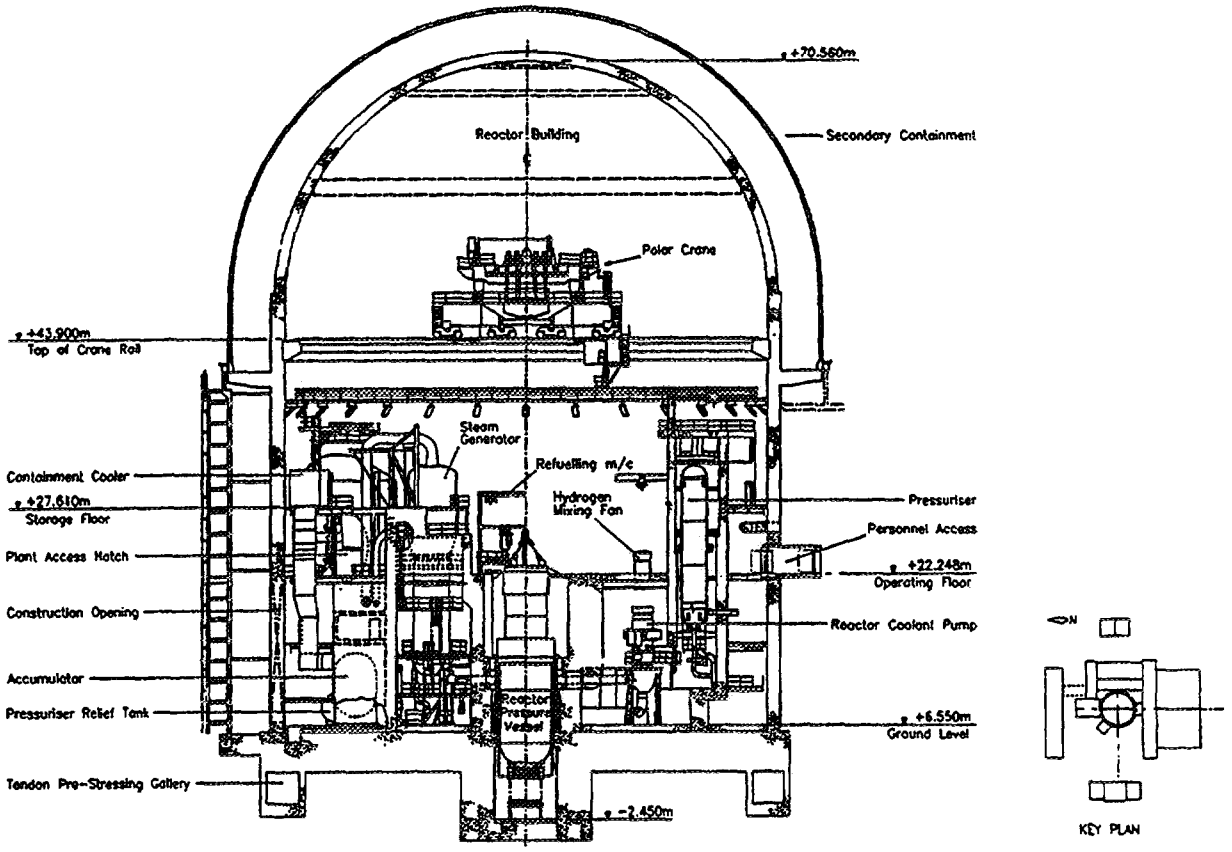


Figure 4.8-10 Sizewell C - Reactor building

A containment isolation system is provided to ensure that if radioactivity is released inside the containment, all penetrations of the containment shell are closed, except those essential for the safety of the reactor. All pipes and ducts passing through the containment wall have at least two isolation valves or barriers. On receipt of a safety injection signal or a manual signal from the control room, all the appropriate penetration isolation devices are automatically closed.

The reactor building contains equipment (eg hydrogen mixing fans, hydrogen recombiners, spray system) to mitigate the consequences of a severe accident (such as a postulated large Loss-of-Coolant-Accident (LOCA) or large steam line break) and is designed to withstand the internal transient effects (pressure, temperature, radiation).

The primary containment is also designed to contain the radioactive products from such postulated major reactor faults. It serves to protect the RCS, including the reactor vessel, steam generators, pressurizer, and reactor coolant pumps etc, from external hazards. Reinforced concrete internal structures support the reactor coolant within the containment. Structures embedded in these are designed to restrain the plant and prevent damage in the event of an earthquake and limit damage due to postulated failure of major high-pressure pipes. The concrete internal structures also function as radiation shields to shield operating and maintenance staff from excessive radiation exposure from the RCS components. The containment internal height is 64 m, its internal diameter is 45.7m, and its wall thickness is 1.3 m; supported within it is a seismically qualified polar crane of 260 tonnes lifting capacity.

Except where it abuts the auxiliary building and fuel building, the primary containment is enclosed by a reinforced concrete (secondary) enclosure building which provides for collection and filtration of leakages from the primary containment. This enclosure building and boundaries of the auxiliary and fuel buildings (to which it is sealed) form a secondary containment enclosing the primary containment structure.

In the event of a LOCA inside the primary containment, or leakage of radioactively contaminated fluid, the normal HVAC systems of the secondary containment structures are suspended and an emergency exhaust system brought into action which creates sub-atmospheric conditions inside the secondary containment and boundary structures, with the filtered exhaust released to atmosphere through a chimney stack.

The reactor building has been designated as a Safety Category 1 and Seismic Category 1 structure and designed to the requirements of Division 2 of the American Society of Mechanical Engineers (ASME) III code. In addition, an ultimate internal pressure nonliner analysis was carried out to determine the ultimate resistance of the primary containment to hypothetical internal pressurization. The design was required to ensure that the ultimate resistance to internal pressure exceeded the design pressure by at least a factor of 2.0, with the strain in prestressing tendons not exceeding one percent.

4.8.7.3 Containment

The containment is essentially a portion of the reactor building, and its description has therefore been merged with that of the latter (Section 4.8.7.2).

4.8.7.4 Turbine building

The turbine hall is a reinforced concrete and steel framed structure. The turbine house comprises two parts: the mechanical annex and the turbine hall. The mechanical annex (which is adjacent to the reactor auxiliary building) houses the main steam and feed pipework, the main boiler feed pumps, the deaerators, the access to the control center and the site incident team assembly room. The annex also forms the main communication access for cableways between the power plant and the control building. The turbine hall contains the main power plant and associated ancillaries.

4.8.7.5 Other buildings

The auxiliary building

The auxiliary building houses engineered safety features and auxiliary systems. The building interfaces with approximately 180^{o} of the reactor building

The control building

The control building is adjacent to the auxiliary building and the turbine house mechanical annex, thus providing convenient access between buildings and minimising cable runs.

The fuel building

The fuel building is immediately adjacent to the reactor building in order to facilitate passage of fuel between the refuelling cavity, in the reactor building, and the fuel storage pond.

The radioactive waste process and storage building

The radioactive waste process and storage building is separated from the main power block but is connected by tunnel to the auxiliary building. The tunnel is divided into a shielded route for active pipework with the remainder used for normal servicing.

4.8.8 Technical data

General plant data		
Power plant output, gross	1388	MWe
Power plant output, net	1307	MWe
Reactor thermal output	3581	MWt
Power plant efficiency, net	36.3	%
Cooling water temperature		°C
Nuclear steam supply system		
Number of coolant loops	4	
Primary circuit volume, including pressuriser		m^3
Steam flow rate at nominal conditions	2000	kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	285/	°C/MPa
Feedwater temperature/pressure	227/	°C/MPa
Reactor coolant system		
Primary coolant flow rate	18300	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	293	°C
Coolant outlet temperature, at RPV outlet	326	°C
Mean temperature rise across core		°C
Reactor core		
Active core height	3.66	m
Equivalent core diameter	3.37	m
Heat transfer surface in the core		m^2
Fuel inventory		t U
Average linear heat rate	18.7	kW/m
Average fuel power density		kW/kg U
Average core power density (volumetric)		kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		
Fuel material	Sintered UO_2	
Fuel assembly total length	4058	mm
Rod array	square, 17×17	
Number of fuel assemblies	193	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	24	
Number of spacers	8	
Enrichment (range) of first core	2.1-3.1	Wt%
Enrichment of reload fuel at equilibrium core	4.5	Wt%
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel	50 000	MWd/t
Cladding tube material	Zr 4	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly		kg
Active length of fuel rods		mm
Burnable absorber, strategy/material		
Number of control rods	53	
Absorber rods per control assembly	24	
Absorber material	Ag-In-Cd	
Drive mechanism	Magnetic jack	
Positioning rate		mm/min
Soluble neutron absorber		
Reactor pressure vessel		
Cylindrical shell inner diameter	4390	mm
Wall thickness of cylindrical shell	215	mm
Total height	13 550	mm
Base material: cylindrical shell	SA508	
RPV head	SA508	
liner	Stainless steel	
Design pressure/temperature	17.2/	MPa/°C
Transport weight (lower part)		t
RPV head		t

Steam generators

Type	U-tube heat exchanger	
Number	4	
Heat transfer surface	5110	m^2
Number of heat exchanger tubes	5626	
Tube dimensions	17.48	mm
Maximum outer diameter	4520	mm
Total height	20600	mm
Transport weight		t
Shell and tube sheet material	SA508	
Tube material	Inconel 690	

Reactor coolant pump

Type	Single-stage, centrifugal pump	
Number	4	
Design pressure/temperature		MPa/°C
Design flow rate (at operating conditions)	4570	kg/s
Pump head		
Power demand at coupling, cold/hot		kW
Pump casing material	stainless steel	
Pump speed	1500	rpm

Pressuriser

Total volume	51	m^3
Steam volume: full power/zero power		m^3
Design pressure/temperature	17.2/	MPa/°C
Heating power of the heater rods	1800	kW
Number of heater rods	78	
Inner diameter	2130	mm
Total height	16100	mm
Material	SA508	
Transport weight		t

Pressuriser relief tank (if applicable)

Not applicable

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	prestressed concrete	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	45.7/64	m
Free volume	90 000	m^3
Design pressure/temperature (DBEs)	340/	kPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate	<0.1	vol%/day
Is secondary containment provided?	yes, reinforced concrete	

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage	23.5/400	kV
	rated capacity	2×800	MVA
Plant transformers,	rated voltage	23.5/11	kV
	rated capacity		MVA
Start-up transformer	rated voltage		kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)			
Number of low voltage busbar systems			
Standby diesel generating units: number		4	
	rated power	8	MW
Number of diesel-backed busbar systems			
Voltage level of these			V ac
Number of DC distributions			
Voltage level of these			V dc
Number of battery-backed busbar systems			
Voltage level of these			V ac

Turbine plant

Number of turbines per reactor	2	
Type of turbine(s)	axial/tandem	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/ LP	
Turbine speed	3000	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	6.68/282	MPa/°C

Generator

Type	3 phase, turbogenerator	
Rated power	2×750	MVA
Active power		MW
Voltage	23.5	kV
Frequency	50	Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type	
Number of tubes	
Heat transfer area	m^2
Cooling water flow rate	m^3/s
Cooling water temperature	°C
Condenser pressure	hPa

Condensate pumps

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

Condensate clean-up system

Full flow/part flow	
Filter type	

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/ °C

Feedwater pumps

Number	
Flow rate	kg/s
Pump head	
Feedwater temperature	°C
Pump speed	rpm

Condensate and feedwater heaters

Number of heating stages	
Redundancies	

4.8.9 Project status and planned schedule

The Sizewell C design is based on the design, construction, commissioning, licensing and operation of the Sizewell B power plant which commenced commercial operation in 1995

The design is state-of-the-art and besides meeting the stringent UK licensing requirements, it can be demonstrated that the design would also be licensable in the US. This is the case because the plant was developed from a US NRC licensed design (i.e., SNUPPS, which was built at both Callaway and Wolf Creek).

The enhancement of the design to meet the more demanding UK design criteria led to a design which anticipated the top level criteria of the EPRI Utility Requirements Documents. The Sizewell C PWR design therefore meets the rigorous standards for Advanced Designs; has already completed its design and licensing; and has a first unit in operation.

The Sizewell C design has been proposed as the basis for future construction in the UK, but at present there are no plans for construction of new nuclear plants in the UK. The design has also formed the basis for a recent overseas bid by Westinghouse/Nuclear Electric.

References

No information provided.

4.9 SWR 1000, SIEMENS, GERMANY

4.9.1 Introduction

In 1992, German utilities awarded Siemens a contract to develop a new BWR nuclear power plant using passive safety features, and together with the utilities Siemens started the development work on the new SWR (Siedewasserreaktor = BWR) with a net capacity of 750 MWe. In the conceptual phase that lasted from February 1992 until September 1993, priority was given to developing passive safety features to replace or supplement active features. During the subsequent consolidation phase, the power level was increased to approximately 1000 MWe, and this design concept is designated the SWR 1000.

The main goal for this advanced BWR is compensation of active safety features used in current designs by passive safety features enabling.

- Reliable control of the various design basis accidents
- Low probability of beyond design basis accidents (core melt frequency)
- Limitation of the impact of core melt accidents to the plant and the environment
- High plant availability
- Economic viability of the concept.

The adoption of passive safety systems requires a lot of engineering efforts, planning and layout work to modify the current design due to the new rules, regulations and requirements. The passive safety features replacing and/or additioning the previously redundant active safety systems must be capable to ensure reliable operation and accident control.

The new features mainly fulfill their duty by physical phenomena such as gravity, natural convection, evaporation or inherent capability of the respective physical phenomena (i. e., temperature coefficient).

At the end of the conceptual phase, it was decided that the new requirements for the advanced SWR, especially economic aspects, motivated a concept with an increased power output:

- Thermal reactor power 2778 MW
- Electric power output (net) 977 MW

Various features have been modified compared with the existing designs:

- The power density of the reactor core is reduced.
- Large water storage capacities inside and outside the reactor containment enable long grace periods and prevent operator interference especially during accidents.
- For transients as well as for accident control, an emergency condenser and an additional containment cooling condenser enable passive decay heat removal from the core and from the containment, respectively.
- The safety functions reactor scram, containment isolation and primary systems depressurization are backed up by passive systems.
- In case of the occurence of a postulated core melt accident the installed cooling capacity is sufficiently conservative to prevent an impact to the structures inside the reactor system or to the environment.
- Despite the introduction of passive safety features for accident control the operating experience gained from current BWR plants constitutes the basis for the new concept.

The new safety features have to be qualified by extensive tests to identify and verify potential weak points in the components and systems design.

This challenge of the advanced reactor development is the real step into the next generation of nuclear power plants.

4.9.2 Description of the nuclear systems

4.9.2.1 Primary circuit and its main characteristics

The nuclear steam supply system is located inside a concrete containment with steel liner and is surrounded by the reactor building.

The main dimensions (22.8 m long and 7.0 m in inner diameter) of the reactor pressure vessel (RPV) correspond to those of the reactor pressure vessel of the 1260 MWe Kruemmel Nuclear Power Plant (KKK). This means that a comparatively large water inventory is available above the core.

Steam is conveyed to the turbine by 3 main steam lines and the feedwater is supplied to the reactor in 2 feedwater lines.

Each of the three main steam lines is equipped with three isolation valves of which two are system fluid- operated and one is motor-operated. One of the fluid-operated valves is located inside the containment, and the other outside it, together with the motor-operated valve.

The two feedwater lines are provided with three check valves as isolation valves, one of them is of diverse design. One valve is located inside the containment, while the other two valves are installed outside.

The outer valves of both steam and feedwater lines are located immediately adjacent to the containment.

The design and mode of operation of the pressure relief system, which serves to limit or reduce the pressure arising inside the reactor under abnormal conditions, are basically the same as those of the pressure relief systems installed in operating BWR plants. For pressure relief or depressurization, the system uses eight safety-relief valves actuated by diverse pilot valves of both active and passive design. A further diverse means for pressure relief is provided by eight rupture disks.

As in the case of large-capacity plants, the reactor is depressurized by discharging steam to the pressure suppression pool via safety-relief valves actuated by diverse pilot valves of both active and passive design. A further diverse means for pressure relief is provided by rupture disks. Once the reactor pressure has been sufficiently reduced, water may flow by gravity into the RPV from an elevated pool, the core flooding pool.

For reducing the buildup of pressure occuring inside the drywell in the event of a loss-of-coolant accident (LOCA), the SWR 1000 has vent pipes located outside the pressure suppression chamber, which terminate at the bottom in inclined ends discharging into the pressure suppression pool water. The routing of the relief lines and the location of the vent pipes outside the suppression chamber air space both constitute a further enhancement of safety provided by the SWR 1000 design concept, since these configurations rule out the possibility of steam being able to leak into the chamber air space.

4.9.2.2 Reactor core and fuel design

The main difference between the SWR 1000 core and previous designs is the reduced core height of 2 800 mm. The reduced core height enables positioning of the core at a lower level inside the reactor pressure vessel, yielding a larger water volume above the core, which supports the accident control. The lowered core also involves reduced length of the control rod drives, which means that the RPV can be located at a lower position inside the reactor building.

The reactor core consists of 648 fuel assemblies of ATRIUM 12 (12-16Q) type. The design of this fuel assembly is based on the ATRIUM 10 (10-9Q) with two additional fuel rod lines keeping the central fuel rod distance and the fuel rod diameter constant. The square water channel occupies the space of 4×4 fuel rods. The fuel channel for the ATRIUM 12 is enlarged correspondingly. The control rod cell is enlarged from 305 to 361 mm due to the ATRIUM 12 geometry. In the core, 157 control elements and 18 lance positions for in-core measurement are foreseen.

The reactor is shut down either by inserting the control rods into the core using the control rod drives, by rapidly inserting all control rods using a hydraulic system or by rapid boron injection into the reactor water. The accumulators of the latter system, filled with a boron solution, perform two functions: first serving as a redundant backup to the water tanks of the scram system for rapid insertion of the control rods and second providing a diverse means for stopping the nuclear fission in the core by injecting the boron solution into the reactor water.

The fuel cycle length is planned for 24 month with a mean reload burn-up of 65 GWd/t. This requires an enrichment in the range of 5.3 Weight% U-235. To ensure adequate shut down reactivity during the entire operation cycle, burnable absorber (Gd_2O_3) is incorporated in the fuel. A low leakage strategy is being applied as fuel management method.

4.9.2.3 Fuel handling and transfer systems

Refuelling will be performed in a similar way as in the currently operating BWRs in Germany. After shutting down the reactor, the containment and the reactor pressure vessel have to be opened. Thereafter, the reactor room and the storage pit are flooded with demineralized water up to the level of the fuel pit. Refuelling can be started after the reactor pressure vessel internals have been removed. By means of the refuelling machine the burned up fuel assemblies are transferred to the level of transport out of the core and transferred to the intermediate fuel storage. After removal of the burned out fuel reshuffling of the remaining fuel assemblies can be performed inside the RPV. The fresh fuel assemblies are by means of the refuelling machine inserted into the reactor core.

For exchange of a control assembly or a control rod guide channel the whole control cell has to be removed. The removal of a control element respectively of a control rod guide channel is performed by means of the refuelling machine and special tools.

The working area of the refuelling machine covers the fuel storage pool, transfer cask pit, lay-down area and flooding room. A telescope auxiliary hoist on the refuelling machine enables the transfer of fresh fuel from the storage room into the fuel pool.

The spent fuel storage in the reactor building shall be sufficient for at least 10 years of operation plus one entire core.

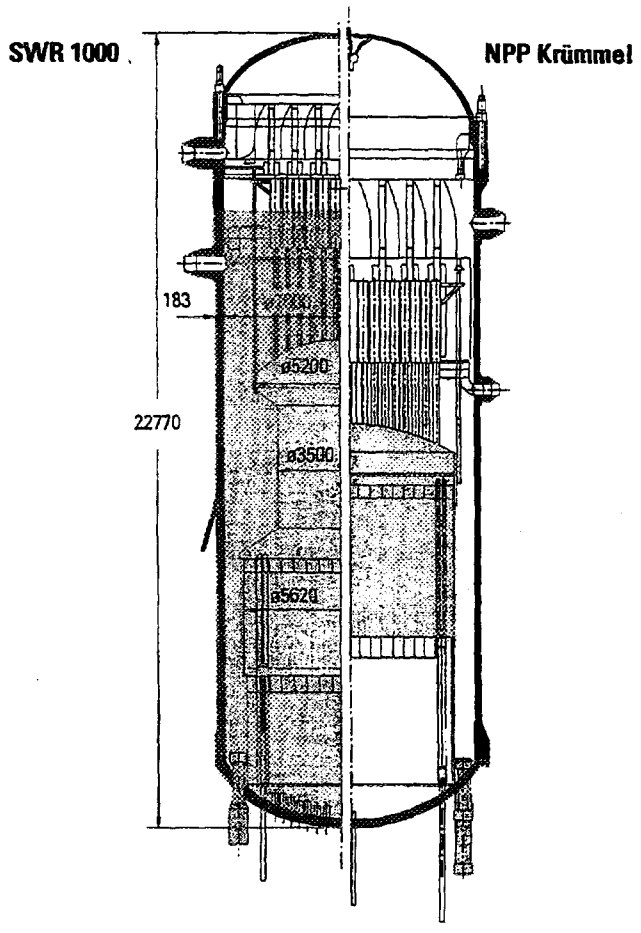


Figure 4.9-1 Reactor vessel comparison between SWR 1000 and NPP Krümmel

4.9.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel (Figure 4.9-1) accommodates the reactor core and the reactor pressure vessel internals. The pressure vessel head is connected to the lower part of the RPV via a flange and fixed by bolts. The complete RPV is supported by brackets/pads in the building's structure. The inner surface of the RPV is lined with stabilized austenitic stainless steel. The pressure vessel is manufactured from 20MnMoNi55 steel. The alloy composition is specified in a very narrow range to optimize the ductility, stresses and resistance against neutron radiation exposure.

Reactor internals

The main differences of the new design in comparison to the current design are the reduced core height and the large water volume above the core. The high water/steam collection room is reduced by a curtailed stack and leads to the larger water volume in the RPV.

All RPV internals, except for the core baffel, are removable to ease the accessibility for in-service inspection.

Reactor recirculation pumps

The six reactor recirculation pumps ensure sufficient cooling of the reactor core. By means of the speed control of the pumps the power output can be controlled. In case of a failure of one pump, the flow through the core can be held constant, by increasing the speed of the remaining pumps. The recirculation pumps are centrifugal pumps driven by wet motors, without mechanical seals and lubricated bearings.

4.9.2.5 Reactor auxiliary systems

Two fully redundant residual heat removal (RHR) trains are provided for removing heat from the reactor after shut down and depressurization. Each train consists of components of the RHR-system, closed cooling water and service water systems. Both RHR trains are equipped with pumps and heat exchangers designed to remove heat from the reactor under low-pressure conditions as well as heat from the containment.

These non-safety-related systems can also be used to supply coolant to the reactor in case of an accident.

The high degree of purity required for the water which circulates through the reactor is ensured by the reactor water cleanup system which is basically of two-train design and continuously removes impurities such as fission and corrosion products from a part flow of reactor water diverted from the reactor. One of the two RHR trains has an additional shutdown filter to enhance reactor water purification while the plant is being shut down. These systems are located entirely inside the containment.

For cooling of the control rod drives and also for supplying makeup water to the scram tanks following a reactor scram, a control rod drive cooling system is provided, which is equipped with two pumps sucking from the condensate storage tank. This system can be used to supply coolant to the RPV at high reactor pressure levels in case of loss of the normal feedwater supply.

The systems and components provided for reactor refuelling, for the storage of new and spent fuel, as well as for handling RPV internals are not different in their design or mode of operation from those installed in current operating SWR plants. The integration of the spent fuel pool cooling and purification systems with the pool is a feature that does deserve mentioning, however. The arrangement of these systems inside the pool dispenses with external connections to the pool below the water level, thereby reducing the potential for leakage from the pool, and eliminates some potentially highly contaminated circuits in the reactor building. The systems for radioactive waste treatment and building ventilation etc., remain basically unchanged.

4.9.2.6 Operating characteristics

The following main performance data will apply for the plant:

- The average annual availability over the plant service life will be > 87%;
- The load-following capability will meet the demands of the German load dispatcher;
- The collective dose of in-house and external personnel should be < 0.5 manSv/a (50 manrem) [as long-term average];
- The duration of scheduled annual refuelling outages shall be < 25 days;
- The number of forced outages (over plant service life) should be < 1/a;
- The average discharge fuel burnup shall be between 50 000 and 65 000 MWd/t U;
- The plant shall permit stretchout operation for70 full-power days;
- The reactor shall permit use of mixed-oxide (MOX) fuel up to 50% without any modifications to the concept, and up to 100% with appropriate modifications; and

4.9.3 Description of turbine generator plant system

4.9.3.1 Turbine generator plant

The turbine is connected to the RPV by three main steam lines. The bypass station that is installed in parallel to the turbine, enables dumping of the entire steam volume generated inside the reactor directly to the condenser. The turbine generator set comprises a 3000 rpm four-casing

single-shaft condensing turbine with directly coupled synchronous alternating-current generator. Both the generator stator and rotor are hydrogen cooled. The main steam from the reactor is admitted to the high-pressure (HP) section of the turbine via combined isolation and control valves.

After undergoing partial expansion in the HP turbine section, the steam passes through a moisture separator without reheater to the three double-flow low-pressure (LP) sections of the turbine. The steam expands in the LP sections through several stages of bladings down to the condenser pressure.

4.9.3.2 Condensate and feedwater systems

The condensate collected in the hotwells of the three condensers is discharged into a common header supplying the main condensate pumps. The condensate passes a demineralizing system before being heated up to the final feedwater temperature in the feedwater heaters. Via two feedwater lines the feedwater is injected into the reactor using speed-controlled feedwater pumps.

4.9.3.3 Auxiliary systems

No information provided.

4.9.4 Instrumentation and control systems

4.9.4.1 Design concepts including control room

The SWR 1000 will use a digital I and C system, covering the following areas:

- operational I and C;
- safety-related I and C;
- process control systems.

The operational I and C systems encompass:

- the process control system including main control room;
- automatic systems including protection equipment and detectors in the plant; and
- bus systems

which are necessary for normal operation (power generation and shut down).

The safety-related I and C protects the human beings and the environment against radioactivity release. During normal operation the safety I and C has no function, however during upset conditions the safety I and C has priority in their functions compared to the operational I and C.

The computerized process information system is responsible for collection of overall plant information. Intelligent information preparation and condensation of the process parameter and performance data are visualized thus enabling a comprehensive survey of operational and safety related functions.

The entire areas of I and C within the plant are connected to each other by redundant bus systems.

Main control room

The entire plant process is controlled via the main instrument panel independent of the mode of operation, i.e., during power operation, as well as start-up and shut-down of the plant. The process information is gathered in a raised instrument panel.

Normal controlling and monitoring results can be adopted from flow diagrams and plasma screens. A touchframe enables manual process interference performed with finger movement on the screen by infrared light.

The information panel consists of colour graphic screens, which are linked to the respective working areas at the main control panel. The colour graphic screens are connected with the process information system.

Emergency control panel

An emergency control panel will be located in one of the emergency power supply buildings. In case of an accident, impact from outside and non-availability of the main control room, shut-down of the reactor and decay heat removal can be initiated from the emergency control panel as far as it was not yet initialized by the automatic protection system.

The emergency control panel is completely independent from the main control room and houses:

- Control and supervision of both decay heat removal systems;
- indication and documentation of special parameters of the incident monitoring system and long range I and C;
- additional indicators and documentation and especially selected process parameter, necessary for function control of passive safety features.

Local panels

Systems, necessary only for part-time operation and without need for direct access from the control room, are operated from local control panels. Examples are the emergency diesel and sampling system.

Monitoring equipment

For reactor control purposes information on the neutron flux density is necessary for a BWR, and the neutron flux level is being directly monitored in the reactor core by means of fixed neutron detectors. The neutron flux monitoring system has the following functions:

- monitoring of the neutron flux density and periodic changes during shut-down conditions, during start-up from cold, subcritical conditions up to 125% of nominal power;
- monitoring of full power core conditions;
- monitoring of the three-dimensional power distribution in the core.

The information is used:

- as input signal for the safety-related I and C, for local core control and for the emergency I and C;
- as input for the nuclear programs of the process computer;
- as information for the operator.

Monitoring of operation parameters for information is essential for the operator as well as for safety-related limit control and control functions.

The proper functioning of the reactor depends on reliable monitoring of the following parameters:

- reactor pressure,
- reactor water level,
- flow rate through the core,
- live steam flow rate, and
- temperature.

For control purposes, redundant measuring channels are foreseen enabling the fullfillment of functions during operation, i. e., shut-down and incidents/accidents control.

4.9.4.2 Reactor protection system and other safety systems

The application of eight pressure pulse transmitters, to initiate reactor scram, containment isolation, RPV depressurization and passive cooling of the reactor core dependent on the water level in the RPV, represents a new feature of the design.

The purpose of the safety-related I and C is to monitor and control important process parameters and actuate safety systems automatically to protect the nuclear power plant and to prevent radiological impact to the environment. The process parameters important for the safety functions are selected and combined to trip signals based on accident analyses. The safety systems of the SWR 1000 consist of both active and passive safety features, and the latter will normally be actuated directly by process parameters without signal from the safety-related I and C.

The safety-related I and C generally detects and records accidents and/or incidents, and initiates active safety systems. In case of non-availability or disturbances in active safety systems, the passive features take over to bring the nuclear power plant to safe condition. All safety features together apply to the deterministic safety concept as it is required in accordance to the rules.

Basically, the detection of an accident is being identified by various process parameters, on which the uncertainties and the impact of common cause failures, based on direct measured information, can be governed.

If these requirements can not be fulfilled, a pressure increase inside the reactor vessel and/ or a reactor water water level decrease will lead to initiation of passive safety systems. A redundant design prevents immanent system failures upon loss of one redundancy since another will directly take over. Isolated decouplings are foreseen at connection points between redundancies.

The reactor protection system is mainly self-testing, and failures are indicated automatically. The active safety features are checked periodically. Active components integrated in passive safety features are being tested in periods > 4 years.

During shut-down of the plant and opened reactor pressure vessel, repairs may be carried out in the two redundant safety systems. In such cases the passive safety features will be quite inactive, the functions normally dedicated to the safety systems have to be taken over by operational auxiliary systems.

4.9.5 Electrical systems

4.9.5.1 Operational power supply systems

The auxiliary power supply system and the grid connection correspond with todays well proven technology.

The generator feeds into the 400 kV grid via the main transformer. The auxiliary power necessary for plant operation is extracted between the generator and the main transformer and fed into the 10 kV switchgear system via two auxiliary transformers.

The generator, main transformer and auxiliary transformers are connected via single-phase enclosed generator buses.

A generator breaker is foreseen between the generator and the main transformer to facilitate the extraction of auxiliary power from the generator and/or from the 400 kV interconnected grid, via the main transformer.

Bridging the operating mode from start-up to operation and vice versa is performed by the synchronization of the generator with the grid respectively shut-down of the generator. For further

independent feeding, a grid system with 110 kV (spare connection) has been selected to grant the energy supply during shut-down of the plant and enabling the operation of the main heat sink.

The main transformer is equipped with a tap changer which permits changing the transformer ratio by a factor of +/-5×0,5% under load. This tap changer is used to modify the reactive power supply to the 400 kV grid or to control the voltage level of the auxiliary power supply system.

The auxiliary transformers have to be capable to control the voltage of the auxiliary power system. These transformers have a fixed transformer ratio, which can be varied off-load by a factor of +/-2×2,5%. The two auxiliary transformers are feeding the 10 kV busbars from the lower voltage winding.

The auxiliary system is divided into two trains, each consisting of a 10 kV main distribution and a 690 V and 400 V low voltage distribution.

The auxiliary loads on the two 10 kV trains are balanced as far as possible, by distributing operating components with similar operating characteristics evenly onto the buses:

- medium voltage motors
- low voltage transformers for supply of the 690 V and for 400 V distributor
- 690 V emergency train for the two redundant decay heat removal systems

The 400 V normal power supply system includes possible re-connections to increase the power supply availability.

4.9.5.2 Safety-related systems

The redundancies are protected within the system and the reactor plant against failure induced events, and each sub-system is protected against the other by physical separation. To protect the redundancies against external impacts, the systems are located in the two emergency diesel buildings, which are protected against external impacts. In each redundancy, a control panel is included, from which the actual plant status can be assessed, and manual actions to shut down the plant, pressure reduction, decay heat transfer can be initiated. A battery-backed energy supply system is incorporated in each of the redundancies. For function control of the passive safety features relevant process parameters are controlled and recorded. For these functions a separate battery-backed energy supply system is installed in the switchgear building.

In the event of loss of auxiliary power supply, safe shut-down of the reactor plant requires start or connection of certain consumers and others to remain in operation. Regarding power supply to safety-related systems a distinction is therefore made between two categories:

- Loads that may be de-energized for a period of 20 s during start-up of the diesel generating units, are connected to the AC power distributions in the emergency power supply system.
- Loads, which can not accept an interruption in operation or which have to be started immediately, are connected to the secured power supply system. They are fed by the 220 V DC system or via a static DC-AC inverter and the connected 400 V distribution.

With respect to accident control, the following process systems are available, besides passive safety features:

- 2×100% decay heat removal systems
- 2×100% additional systems with diverse emergency power consuming components.

For emergency power supply of these systems a two train emergency system is installed. Each train consists of:

a) An emergency diesel generator unit that takes over power supply automatically (within 20 s) in the event of loss of auxiliary power.

b) 690 V AC power distributions for emergency supplied components with a power demand of 130 - 400 kW.

c) Low voltage transformer with 400 V three phase AC distribution for loads < 130 kW.

d) 220 V DC power supply system with battery and charging rectifier.

e) Static power inverter 220 V =/230 V".

f) DC/DC converter for +/- 24 V-loads.

Each 690 V emergency power distribution has the following feeding possibilities

1) via a connection to the same auxiliary power supply train (normal operation)

2) from a diesel generator unit.

4.9.6 Safety concept

4.9.6.1 Safety requirements and design philosophy

The safety concept of the SWR 1000 is based on two fundamentals:

1) accident prevention; and

2) accident mitigation

Innovative measures are introduced in the accident mitigation concept.

With respect to prevention of accidents, all proven measures of current safety concepts assuring reliable operation are incorporated in the SWR 1000 plant design as demonstrated by the following examples:

- Systems and components are designed with conservative margins.
- Application of strict rules to eliminate failure of major components (the German *Basissicherheitskonzept)* for design, material selection, manufacturing and testing.
- Application of proven QA programmes considering the experience gained from design up to commissioning.
- Operation surveillance by control systems and indication systems.
- Use of proven mechanical and electrical components and I and C.
- In-service inspection.
- Preventive maintenance.
- Use of the redundancy concept for operating systems to increase the reliability.

All requirements mentioned in accident prevention rules for PWRs in Germany have been considered, as far as they would be applicable for a BWR. Further precautions against hypothetical core melt accidents have been made to limit the impact and the consequences for the plant.

The new concept takes advantages from the reduced power density in the core and the large water reservoires in the RPV, condensation chamber and in the flooding pool. On the other hand it takes advantage from the active and passive system functions, which are redundant and diverse relative to each other. This leads to the three step safety concept delineated below.

On the first level, the necessary functions for the control of accidents are covered by active emergency power supplied systems, initiated by I and C signals.

At the second level, all safety functions are covered by additional and diverse passive systems. These systems fulfill their functions without I and C input and electric power supply. They are initiated by the direct impact of various process parameters. The design is redundant and diverse to the measures of the first level.

At the third level, additional independent diverse and passive measures are installed to cope with postulated, hypothetical core melt accidents, to keep the melt inside the reactor pressure vessel (RPV). For this reason, water from the flooding pool is injected into the pressure chamber (around the RPV) for effective cooling of the RPV from the outside.

The planned active and passive safety systems are described in more detail in the following section.

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

The necessary safety functions for accident control are ensured by redundant, active systems. The design of the active systems takes the existence of additional passive systems to realize the deterministic redundancy concept into account in connection with the single failure strategy.

For fast shut-down of the reactor, a scram system with two independent trains activates the control rods, each train covering 50% of the rods. Each train has a normal water storage tank and a borated water storage tank, with two functions: the energy for insertion of the control rods is delivered from the system, and the boric acid injection into the core is considered as a diverse shut-down system.

In addition, the control rods can be moved into the core using electric drives, which are connected to the back-up energy supply system.

The main steam and feedwater lines are provided with three normally open isolation valves. One valve is located inside the containment, the other two outside the containment. One valve is diverse to the other two. Two of the valves are medium operated, the third has a separate actuator.

The I and C-system generates the signals for closing of the main steam valves via electro-magnetic pilot valves, which operate in fail-safe mode. The check valves in the feedwater lines do not need any control signal.

Following an isolation from the main heat sink, the eight safety and relief valves will blow steam down into the condensation chamber with its pressure surpression system to prevent excessive pressure in the RPV. The valves are medium operated.

The above-mentioned eight safety and relief valves also accommodate the function of relieving the pressure in the RPV in connection with a loss of coolant accident. To keep the main valves open during decreased RPV pressure they are mechanically locked.

With respect to the core flooding functions, two existing operational systems are used, for the high and low pressure range:

- High pressure injection with the control rod flushing system; and
- Low pressure injection with the residual heat removal system.

Active components of the two train systems feed all control rod drives with flushing and cooling water. This emergency power secured operating system should be used as the high pressure injection system during unavailability of the feedwater system. The discharge capacity planned for normal operation ensures filling of the RPV after a reactor scram to keep the water level in the RPV above the filling mark for pressure relief activation. Feeding can be performed with one train.

The pump design permits injection also during reduced RPV pressure, therefore a continuous transition for further low pressure feeding with the residual heat removal system is possible.

The two train emergency power secured residual heat removal system with primarily operating functions will also be used as a low pressure core flooding system during loss of coolant accidents. The water is transferred from the condensation chamber in the RPV under pressure < 1 MPa (10 bar). The feeding capacity of one train is sufficient for this operating mode. During flooding, heat is transferred from the containment to the ultimate heat sink.

Under accident conditions (i. e., failure of the main heat sink, loss of coolant accident), which lead to a separation from the main heat sink, the residual heat removal cooling system in connection with the condensation chamber of the wetwell (flooding pit/water cooling system) is used for heat transfer from the containment to an ultimate heat sink.

Active and passive safety features constitute together the safety concept of the SWR 1000. Assuming non-availability of all active systems during an accident, accident control is exclusively performed by the passive systems that are shown in Figure 4.9-2. These systems work without electric energy and I and C signals, and they are of redundant design and diverse redundant to the active systems.

For passive initiation of reactor scram, isolation of main steam lines and automatic depressurization of the RPV, a passive pulse transmitter is installed. The transmitter is working very much like a small emergency condenser. In the case of a reduced RPV water level, steam enters the transmitter and heat is transferred to a secondary medium that, in turn, will actuate e.g. pilot valves. This leads to a completely passive triggered chain.

Activation of scram and main steam line isolation is initiated by 4 redundant pulse transmitters located at reactor water level LT1. Automatic pressure relief, in connection with a LOCA, will be initiated from 4 pulse transmitters, which are installed at a reactor water level LT2.

In the scram system, diverse passive pilot valves are installed in addition to the magnetic pilot valves in each train, for passive initiation of scram; the passive pilot valves are controlled by passive pulse sensors, that means without I and C.

Pos.		Number
1	Emergency condenser	4
2	Safety-relief valve	8
3	Spring-loaded pilot valve	8
4	Diaphragm pilot valve	8
5	Passive pressure pulse transmitter	2 x 4
6	Rupture disk	8
7	Flooding line	4
8	Containment cooling condenser	4
9	Core flooding pool	
10	Pressure suppression pool	
11	Vent pipes	15
12	Scram system	2

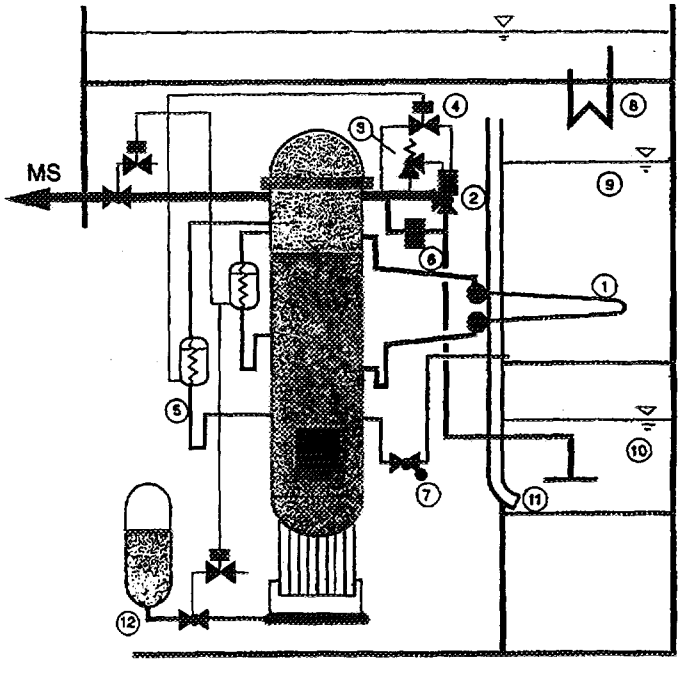


Figure 4.9-2 SWR 1000 - Passive systems

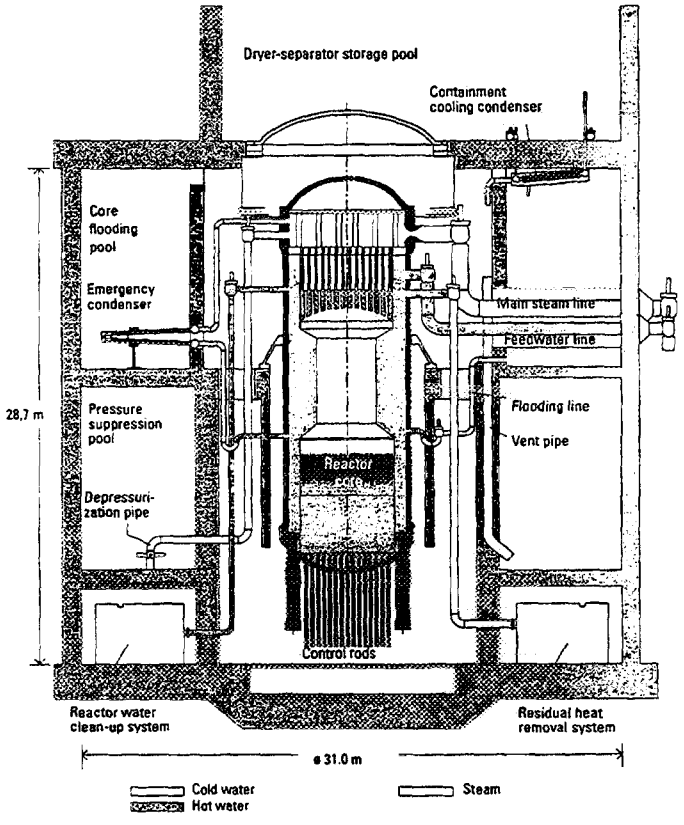


Figure 4.9-3 SWR 1000 - Containment arrangement

The passive trail of the RPV pressure limitation system includes redundant and diverse features. In addition to the active magnetic pilot valves for each of the eight safety and relief valves, one extra passive spring-loaded pilot valve is being planned. Additionally to the passive control of the safety and relief valves redundant and diverse rupture discs (8) are installed.

For pressure relief, which is only necessary during a LOCA, each of the safety and relief valves will be equipped with a diaphragm pilot valve actuated by a passive pulse transmitter.

The emergency condenser which consists of 4×25% units, has to fulfill two functions:

- To release the decay heat to the flooding pool water; and
- to recycle the condensate back into the RPV.

The emergency condenser itself is a fully passive component of particular importance, particularly for controlling transients. It is located in the core flooding pool and is connected to the RPV by non-isolatable steam discharge and return lines. It starts operation upon dropping water level in the RPV.

For passive heat transfer from the containment, a separate containment cooling condenser will be installed, divided into 4×25% units. The steam released into the containment by a LOCA or evaporation of water from the core flooding pool condenses in the containment cooling condenser and is recycled into the flooding pool. The condenser works by natural circulation.

The basic arrangement of the containment is depicted in Figure 4.9-3.

4.9.6.3 Severe accidents (Beyond design basis accidents)

Severe accidents have a very low probability of occurence. The combined functions of passive safety equipment and active non-safety-related systems result in an estimated probability of

occurence, which is two magnitudes lower than that of current reactors, based on the initiating events taking place during power operation addressed above (transients and LOCAs).

Nevertheless, it is postulated that a core melt can occur. In such a case, flooding lines from the core flooding pool will open to supply water to the outside of the RPV, so that the melt can be cooled while still inside the vessel. This ensures retention of most of the radioactive aerosols inside the containment.

The release of the most significant isotope, Cs-137, which is estimated to approximately 10^{11} Bq is far below the level of 10^{14} Bq, which has been internationally discussed and accepted. The containment is inerted and the release of hydrogen would not lead to either combustion processes or an explosion. The design of the containment is such as to preclude loss of containment integrity under such pressure loadings.

4.9.7 Plant layout

4.9.7.1 Buildings and structures, including plot plan

A plot plan of the plant is shown in Figure 4.9-4. In the organization of the buildings relative to each other a differentiation is made between site-independent buildings, such as: Containment (UJA); Reactor building (UJB);Off-air stack (UKH); Turbine building (UMA); Auxiliary system building (UKA); Service building (UKB); and Switchgear building (UBA), and site dependent buildings, such as cooling water system and conventional auxiliary building.

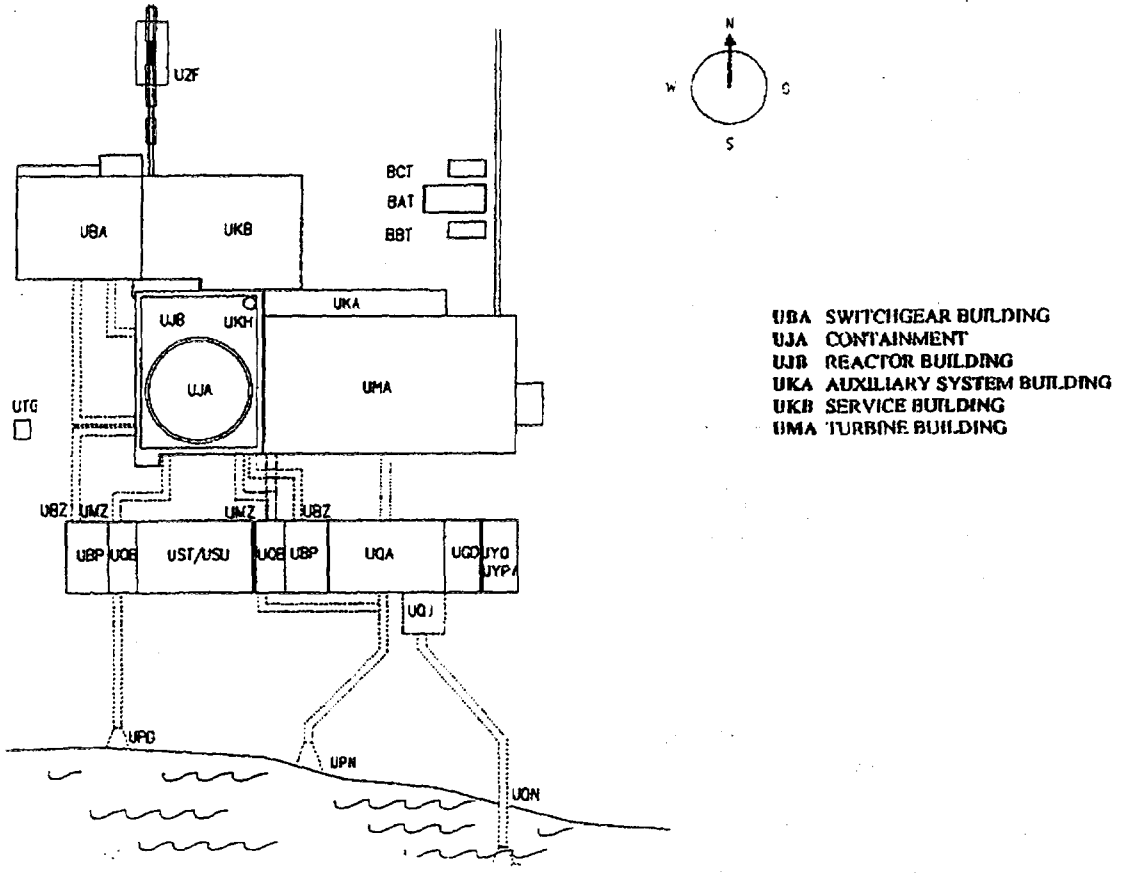


Figure 4.9-4 SWR 1000 - Plot plan

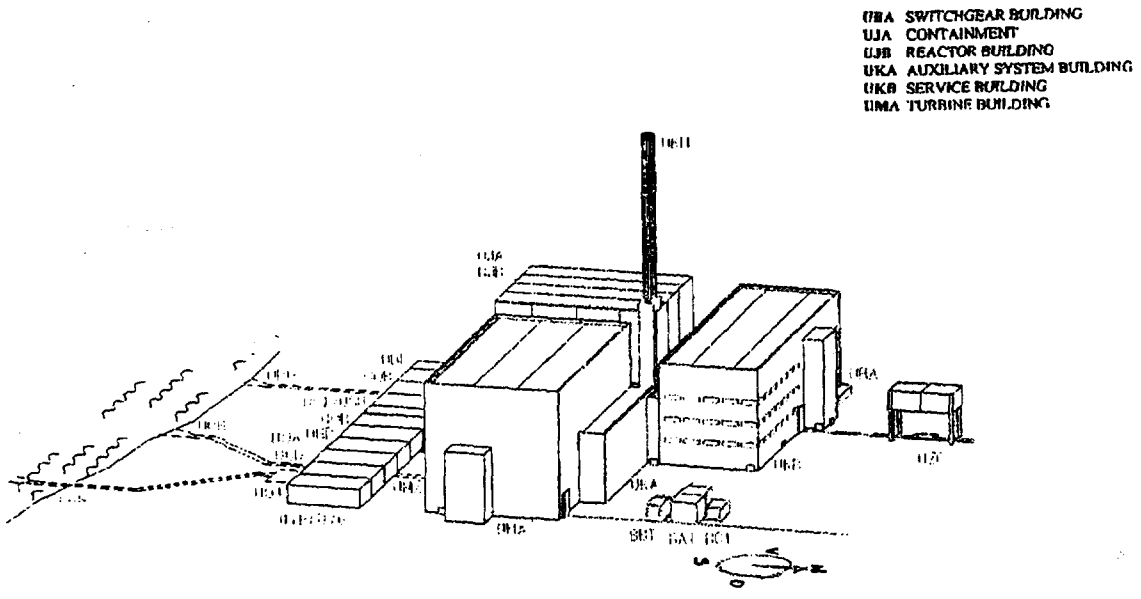


Figure 4.9-5 SWR 1000 - Building arrangement; view from north-east

The overview of the building arrangement (Figure 4.9-5) gives further information on the location of the different buildings.

The buildings have been organized compactly taking into consideration the critical path for the construction schedule, the accessibility and systems oriented interfaces to assure optimized construction and later on short distances between systems for the operating staff.

The central unit consists of the reactor building with integrated stack and the turbine building with integrated auxiliary building. Adjacent to these buildings the reactor auxiliary building with the entrance to the controlled area and the switchgear building with the main control room is located. The central stair-case and the elevators permit a centralized personnel entrance and exit.

At a certain distance from the power plant buildings and the cooling water entrance, several other buildings are located housing the pump station, emergency diesel and the auxiliary systems. Access by the staff during plant operation is possible via a personnel bridge from the social area of the auxiliary building.

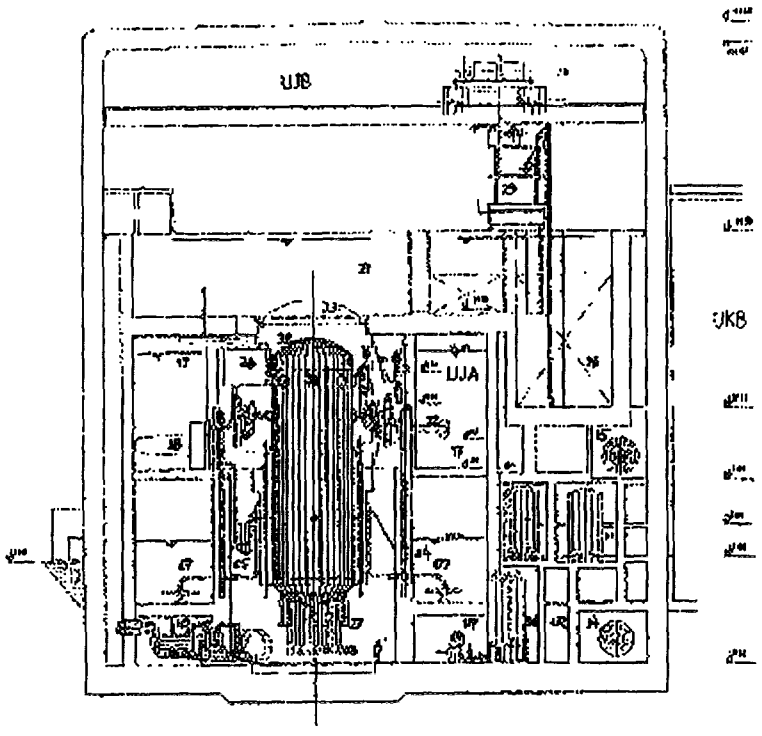
The ducts of the ventilation and air conditioning systems are running through the turbine building. The cable trains are installed in the ground or in cable channels. Crosspoints of safety-related important trains are avoided.

The access to the NPP is only possible via one entrance. All other doors, enabling heavy transports and access of the fire brigade, are normally closed. These entrances are only opened on controlled request.

Design requirements (Earthquake, Aircraft crash, Explosion pressure wave, Internal hazards, Physical separation aspects, and Radiation protection aspects (accessibility, shielding, ventilation) have not been described.

4.9.7.2 Reactor building

The reactor building accommodates all systems and components necessary for reactor operation and reactor protection.



UJA CONTAINMENT

01 Reactor pressure vessel
02 Pressure suppression chamber
03 Control rod drives
04 Vertical discharge
05 RWCU system filter
07 SRV discharge line
09 Scram tank valves
10 RHR system, pump & heat exchanger
16 Main stem line isolation valve
17 Core flooding pool
18 Emergency condenser
19 Relief valve
20 Feedwater isolation valve
22 Pipe duct
26 Containment cooling condenser
27 Recirculation pumps
28 Reactor operating floor
32 Vessel-to-drywell seal

UJB REACTOR BUILDING

02 Corridor
06 Scram and boron injection tanks
07 Concentrated waste tank
14 Condensate demineralizing system; backwash tank
15 Condensatestorage tank
26 Fuel pool
27 Dryer-separator storage pool
28 Reactor building crane
29 Refuelling platform
33 Drywell head

Figure 4.9-6 SWR 1000 - Reactor building; longitudinal section A-A

The complete fuel storage and handling facilities are located in the protected area. The other compartments not occupied by the primary system are being used for installation of reactor auxiliary systems. Short cable and pipe routes are chosen. Redundancy is reached by physical separation of safety-related systems.

The compact arrangement of systems and components enables optimal radiation protection and emergency protection.

The reactor building (Figure 4.9-6) accommodates the following installations:

- Containment
- Fuel handling and storage facilities
- Scram tanks and pumps for control element drive flushing
- Condensate tank
- Ventilation and air conditioning systems for the controlled area
- Radioactive waste concentrate tank
- Off-gas system (delay beds)
- Evaporation plant
- Evaporator feed tank
- Active waste collection tank
- Condensate flushing tank of the main condensate system.

The dimensions of the reactor building are dictated by the size of the RPV, and the condensation chamber, as well as by the flooding pool, the laydown area and the fuel storage pool.

4.9.7.3 Containment

The pressure retaining components of the reactor system, mainly the RPV as well as the direct connected systems up to the second isolation valve, are installed inside the containment.

Some of the auxiliary systems are also located inside the containment.

Plant components

The plant components are located below the condensation chamber. In this area the following components are installed:

- Residual heat removal pumps and coolers
- Fast opening tank isolation valves for the control rod drives
- Reactor water cleanup system pumps
- Reactor water cleaning coolers and filters.

The heat transfer out of the compartment is achieved by means of gas circulation coolers. The compartments are vented and can be made accessible and entered by the staff under consideration of radiological requirements.

Drywell

This room accommodates the flooding pool and inner cylinder, and houses the following components:

- RPV
- Feedwater and main steam lines and isolation valves
- Safety and relief valves with exhaust pipe
- First isolation valves of the residual heat removal system
- Control rod drives and ring pipes of the scram system
- Recirculation pumps
- Emergency condenser with condensate and steam pipes
- Containment cooling condenser
- Reactor water purification clean-up filter
- Passive pulse transmitters
- Flooding tubes
- Measuring pipes
- Vent pipes
- Air circulation units (3×50%).

Wetwell

The wetwell is filled with water up to a level of about 40%. For pressure suppression during incidents the released steam is injected into the water where it condenses. The vent pipes and the quencher tubes enter the water in the condensation chamber.

Containment integrity and corium retention

No description provided.

4.9.7.4 Turbine building

The equipment of the water steam cycle including turbine and generator, the pre-heaters and the feedwater system including feedwater tank and feedwater pumps are located in the turbine building. At the lowest floor the main cooling water pipes enter the building. The condensers are located transversal and occupy the room up to the turbine operating floor. The cooling water inlet and outlet of the condenser are held short.

The last section of the turbine building up to 0.00 m is designed as an air lock; tracks enable connection with the railway system. Near the air lock; a condensate filter system is positioned.

The preheaters are located on the intermediate sealing adjacent to the turbine. The preheaters expand from the floor +/- 0.00 m up to under the operating floor. The vertical preheaters can be transferred to its position with the turbine building crane using erection openings at the operating floor.

The lubrication system for the turbine consisting of an oil tank and pumps is located in a separate against fire protected compartment. At the same level the generator bus leaves the turbine building penetrating the outer wall and connects the generator directly with the main transformer.

On the turbine operating floor only the two condensate storage tanks are installed, the rest remains free and is being used as storage area during maintenance. A turbine building crane enables the transport of equipment. The feedwater and main steam lines enter and leave, respectively, the turbine building at the floor + 8 m.

4.9.7.5 Other buildings

Service building

The tanks being used for waste water storage and treatment are located in the service building. Further design details of that building will be substantiated at a later date.

Auxiliary system building

The work shops, the waste treatment and storage facilities and the central entrance to the controlled area are located by the auxiliary system building. The chosen design leads to short pipe and cable routes to the systems located in the reactor building. The auxiliary system building has 5 floors housing the following systems or parts of them:

- Solid waste treatment
- Decontamination plant
- Fresh and off-air systems
- Hygiene rooms, especially for the controlled area (showers, lavatories, toilets)
- Laboratory
- Hot and cold work shop
- Office and storage rooms

The complete staff movement is being checked and guided. The entrance to the controlled area is located on about 13.30 m. For the transport of the components into the hot work shop the equipment has to pass the airlock.

Switchgear building

The switchgear building accomodates two energy supply units, physically separated each to each other and constitutes the redundancy for the switch gear and I and C-systems. Each part has its own cable, fresh and off-air system.

Each switchgear section has its own independant cable, fresh and off-air, smoke extraction and cable ducts.

The switchgear building has 9 floors; at the lowest level are cable tracks, three floors are occupied by auxiliary power systems as well as by I and C. At the two highest floors, the main control room and the supervisory computer are located with the necessary air conditioning systems.

4.9.8 Technical data

General plant data

Power plant output, gross		MWe
Power plant output, net	977	MWe
Reactor thermal output	2 778	MWt
Power plant efficiency, net	35.2	%
Cooling water temperature		°C

Nuclear steam supply system

Number of coolant loops	1	
Primary circuit volume		m^3
Steam flow rate at nominal conditions	1 475	kg/s
Feedwater flow rate at nominal conditions	1 475	kg/s

Reactor coolant system

Primary coolant flow rate	12 000	kg/s
Reactor operating pressure		MPa
Steam temperature/pressure		°C/MPa
Feedwater temperature		°C
Core coolant inlet temperature		°C
Core coolant outlet temperature		°C
Mean temperature rise across core		°C

Reactor core

Active core height	2.8	m
Equivalent core diameter		m
Heat transfer surface in the core		m^2
Average linear heat rate		kW/m
Fuel weight {U or UO_2?}	127	t U
Average fuel power density		kW/kg U
Average core power density	48	kW/l
Thermal heat flux, F_q		
Enthalpy rise, F_H		

Fuel material	Sintered UO_2	
Fuel (assembly) rod total length		mm
Rod array	12×12-(4×4) in square lattice	
Number of fuel assemblies	648	
Number of fuel rods/assembly	128	
Number of spacers		
Enrichment (range) of first core, average	4,95	Wt%
Enrichment of reload fuel at equilibrium core	~5,3	Wt%
Operating cycle length (fuel cycle length)	-24	months
Average discharge burnup of fuel	65 000	MWd/t
Cladding tube material	annealed, recrystallised Zr 2	
Cladding tube wall thickness		mm
Outer diameter of fuel rods		mm
Fuel channel/box; material	Zr-4	
Overall weight of assembly, including box		kg
Uranium weight/assembly		kg
Active length of fuel rods	2 800	mm
Burnable absorber, strategy/material	Gd_2O_3 in fuel	
Number of control rods	157	
Absorber material		
Drive mechanism	electro-mechanical	
Positioning rate		mm/s
Soluble neutron absorber	Boron	

Reactor pressure vessel

Inner diameter of cylindrical shell	7 000	mm
Wall thickness of cylindrical shell		mm
Total height, inside	22 800	mm
Base material: cylindrical shell & RPV head	20MnMoNi55 steel	
lining	stainless steel	
Design pressure/temperature	8.8/	MPa/°C
Transport weight (lower part)		t
RPV head		t

Reactor recirculation pump

Type	Integral, glandless, centrifugal pump	
Number	6	
Design pressure/temperature		MPa/°C
Design mass flow rate (at operating conditions)	2 000	kg/s
Pump head		MPa
Rated power of pump motor (nominal flow rate)		kW
Pump casing material	as for RPV	
Pump speed (at rated conditions)		*rpm*
Pump inertia		*kg m²*

Primary containment

Type	Pressure-suppression	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)		m
Design pressure/temperature		kPa/°C
Design leakage rate		vol%/day
Is secondary containment provided?		

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure (100 °C)		MW
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage	440/	kV
	rated capacity		MVA
Plant transformers,	rated voltage	/10	kV
	rated capacity		MVA
Start-up transformer	rated voltage	/10	kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)		2	
Number of low voltage busbar systems			
Standby diesel generating units: number			
rated power			MW
Number of diesel-backed busbar systems			
Voltage level of these			V AC
Number of DC distributions			
Voltage level of these			V DC
Number of battery-backed busbar systems			
Voltage level of these			V AC

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	axial, double flow, condensing	
Number of turbine sections per unit (e.g. HP/LP/LP)	HP/3 LP	
Turbine speed	3 000	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	6,7/	MPa/°C

Generator

Type	3-phase, turbo-generator	
Rated power		MVA
Active power		MW
Voltage		24 kV
Frequency	50	Hz
Total generator mass, including exciter		t
Overall length of generator		m

Condenser

Type	shell type	
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate		m^3/s
Cooling water temperature		°C
Condenser pressure		kPa

Condensate pumps

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

Condensate clean-up system

Full flow/part flow	full condensate flow
Filter type	

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/ °C

Feedwater pumps

Number	3 × 50%	
Flow rate		kg/s
Pump head		MPa
Feed pump power		MW
Feedwater temperature (final)		°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages,	low pressure
	high pressure
	feedwater tank

4.9.9 Project status and planned schedule

The contract between Siemens and the German utilities runs for a period of 4 years, from 1995 to 1999. During this period, the necessary R&D activities shall be completed, the basic engineering work performed, and licensing documents prepared. The BWR shall be licensable in Germany in accordance with German rules, regulations and requirements, and also meet the European Utility Requirements. The conceptual design shall be established in early 1997.

Tests have been performed on the emergency condenser and these tests have confirmed the calculated performance; the full cooling capacity was attained earlier than expected and design margins could be quantified.

References

Relevant information is not yet available.

4.10 SYSTEM 80+, ABB CENP, USA

4.10.1 Introduction

The System 80+ Standard Plant Design was developed by incorporating improvements into the System 80 design used for the plants already in operation at the Palo Verde Nuclear Generating Station. To produce a safe and simpler design with greater reliability and enhanced operability, the design followed four sets of requirements and policies: (1) the Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD), which specifies characteristics desired by utilities in future plant designs; (2) the US. Nuclear Regulatory Commission (NRC) Severe Accident Policy, which identifies new safety standards to be applied to future nuclear plant designs; (3) the US. Code of Federal Regulations, Title 10, Part 52 (10 CFR 52), which provides the framework for licensing of new standardized designs[1]; and (4) current US. NRC regulations, supplemented by emerging policy issues, which are summarized in SECY-93-087.

The System 80+ Standard Plant Design meets the stringent design goals in the EPRI ALWR URD Volumes I and II. The design represents an evolutionary advancement over its predecessors, and no prototype testing is required. Further, unproven features that could slow the licensing process and/or construction or could result in unreliable operation have been avoided.

The System 80+ design also complies with the procedural requirements and criteria of NRC regulations including the Three Mile Island requirements codified in 10 CFR 50. In addition, the design addresses all applicable Unresolved Safety Issues (USIs) and the medium- and high-priority Generic Safety Issues (GSIs). Finally, a full Level III Probabilistic Safety Assessment (PSA) has been carried out for the design. The PSA was used as a guiding tool during the design process to produce a more robust design that minimizes the potential for core damage and moderates the severity of a severe accident should one occur. Accordingly, the design meets the NRC Severe Accident Policy.

As required by 10 CFR 52, the scope of the System 80+ Standard Plant Design covers an essentially complete nuclear power plant and includes all structures, systems, and components that can significantly affect safe operation. An overview of the System 80+ design is shown in Figure 4.10-1 on next page.

Features that contribute to the safety improvements of the System 80+ design include:

- Increased reactor core thermal margin.
- Use of a ring-forged reactor pressure vessel with improved material specification affording a low 60 year end-of life RT_{NDT}, virtually eliminating pressurized thermal shock concerns. This feature also results in a significantly reduced number of welds (with resulting reduction in in-service inspection).
- Pressurizer volume is increased by 33% (relative to predecessors).
- Secondary inventory in the steam generators is increased by 25%.
- Thermally treated Inconel 690 tubing is used to extend the life of the steam generators, improve their reliability and decrease the potential need for plugging tubes over the plant life.
- Increases in pressurizer volume and steam generator tubing result in an 8% increase in reactor coolant system inventory.
- Gamma ray monitors, one per steam generator, are incorporated in the steam lines to provide a sensitive and specific indication for primary coolant leakage through steam generators.

[1]The 10 CFR 52 approach provides a process for resolving licensing issues related to the design before any commitment to construction. A utility can reference a Certified Design and apply for a single combined license, authorizing both construction and operation, with assurance that the NRC staff will accept the certified portion of the design without further review. Furthermore, any public hearings undertaken on a specific combined operating license (COL) application would exclude issues related to the certified portion of the standard plant design. This process will allow utilities to plan for new nuclear plants by reducing the uncertainty associated with regulatory delays or design modifications during plant construction and startup.

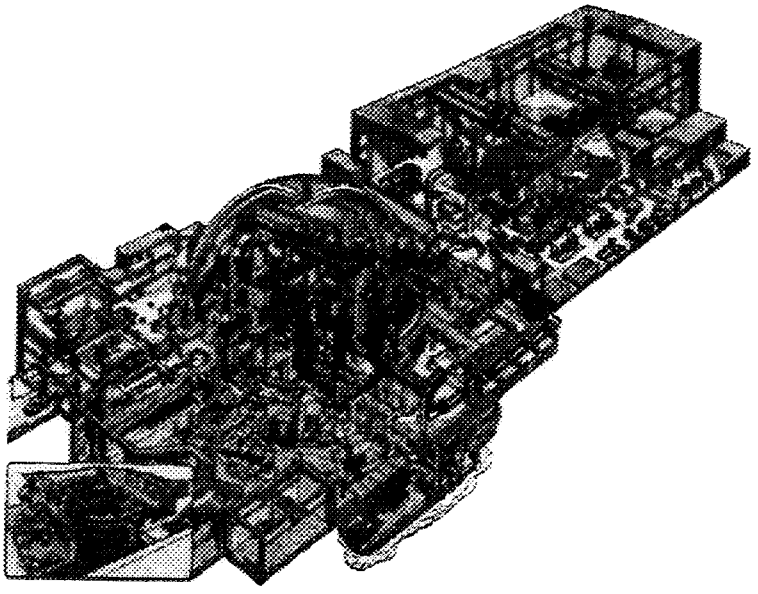


Figure 4.10-1 Overview of System 80+ Design

- A combustion turbine generator provides an alternate source of AC electrical power during loss-of-off-site power and station blackout events.
- A dedicated reactor coolant pump seal injection system has been incorporated.
- A safety depressurization system (SDS) has been added to provide rapid depressurization for severe accident mitigation and for back-up decay heat removal.
- An in-containment refuelling water storage tank (IRWST) acts as a quench tank for the SDS, avoids the need for safety injection recirculation switch-over to the containment sump after a loss-of-coolant accident, and provides a source of water for cavity flooding.
- The cavity flooding system allows for flooding of the reactor cavity in a severe accident.
- A state-of-the-art main control room (Nuplex 80+) uses modern human factors engineering techniques and off-the-shelf digital technology.
- Hard-wired monitoring and component actuation capability to provide further diversity to the plant's already redundant Alternate protection system.
- Additional mechanical redundancy has been provided for the safety injection, emergency feedwater, shutdown cooling, and containment spray systems.
- A large free containment volume provides additional margin against overpressurization and ensures that global hydrogen concentration cannot reach detonable levels during an accident.
- A hydrogen igniter system, in conjunction with hydrogen recombiners, ensures that hydrogen is controlled without global deflagrations.

The System 80+ design incorporates also features to reduce O&M costs, including:

- Use of proven components to enhance reliability and availability.
- Features to expedite maintenance such as adequate access provisions and permanently installed work platforms and handling equipment.
- Reduced maintenance and inspection, e.g., due to elimination of various welds and valves.
- Measures such as material selection, use of shielding, and physical layout to reduce radiation exposure and, therefore, reduce personnel requirements and facilitate maintenance.
- Provisions for one-piece removal of large components in containment.
- Improved fuel handling equipment and facility arrangements to expedite fuel handling and inspection.
- Nuplex 80+ ACC incorporating human factors engineering and digital instrumentation and controls technology to enhance plant operability.

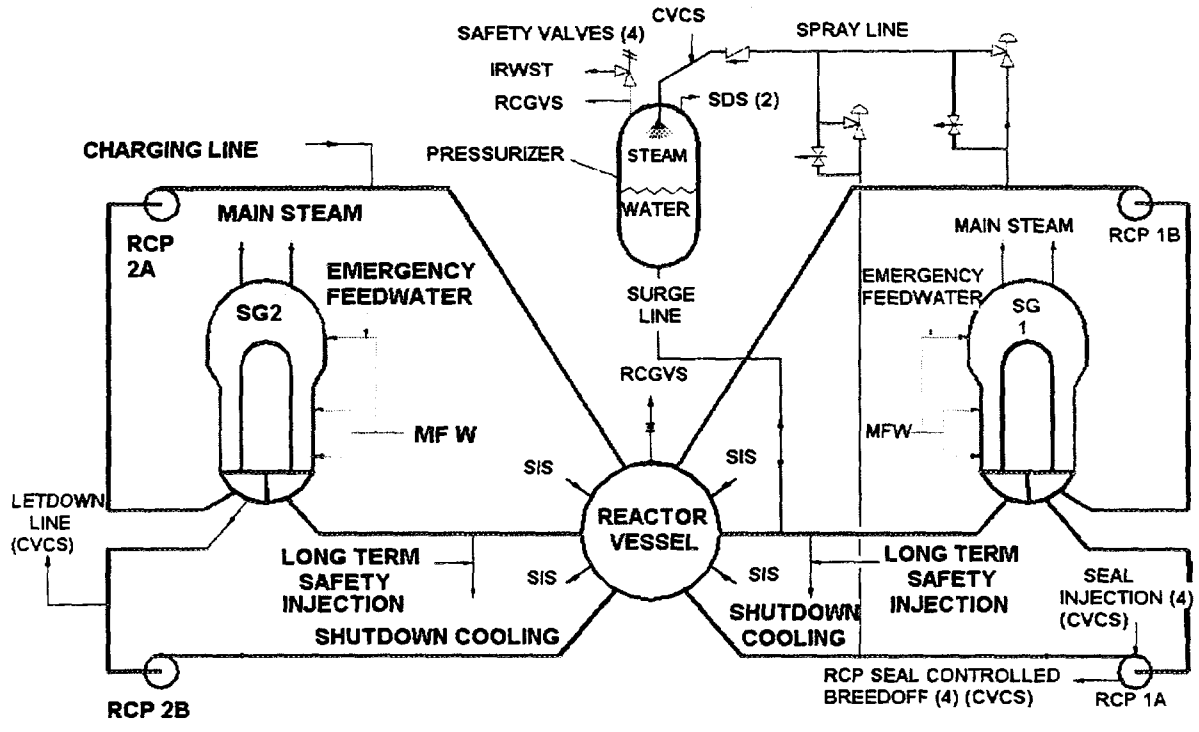


Figure 4.10-2 Reactor coolant system

- Nuplex 80+ features to enhance maintainability such as reduction in the number of sensors, automatic on-line testing, integral test and maintenance panels, use of standardized components, modular equipment design, and modular software coding.
- Provisions to segregate waste streams and reduce amounts of liquid and solid radioactive waste.
- Full-flow, on-line testing capability for fluid-mechanical safety systems.
- Redundancy in various auxiliary systems to permit maintenance while the system is operating.
- Use of an IRWST that results in elimination of valves, piping, heat tracing, and thermal transients associated with an external tank.
- Reduction of piping and plant components exposed to raw water cooling.
- Improved maintainability and operability of heating, ventilation, and air conditioning (HVAC) systems.
- Multiple stud tensioner that permits tensioning/detensioning all reactor vessel studs at once and their simultaneous removal to the storage area.
- Bottom-mounted in-core instrumentation that can be removed/installed in parallel with reactor vessel head removal/installation rather than in series.
- Control element assemblies removal from the core with removal of the upper guide structure.
- Head area cable tray system with integral missile shield that allows simultaneous removal/replacement of cabling in the reactor vessel head area.
- Permanent refuelling pool seal assembly that eliminates installation/removal of the seal.

4.10.2 Description of the nuclear systems

4.10.2.1 Primary circuit and its main characteristics

The nuclear steam supply system (NSSS) is a pressurized water reactor (PWR) with a reactor coolant system (RCS) (Figure 4.10-2) comprising two independent primary coolant loops, a pressurizer connected to one of the loops, two steam generators, four reactor coolant pumps (RCPs) and the auxiliary and safety systems directly related to the NSSS. The NSSS generates approximately 3931 MWt, producing saturated steam at 6.9 MPa (1000 psia).

Each loop consists of a 1065 mm (42 inch) ID outlet pipe, two 760 mm (30 inch) ID inlet pipes, a steam generator and two reactor coolant pumps (RCPs). The RCPs are electric-motor-driven single-stage centrifugal pumps. The RCS operates at a nominal pressure of 15.5 MPa (2250 psia). System pressure is maintained by an electrically heated pressurizer that is connected to one of the loops. Each steam generator is a vertical U-tube heat exchanger used to transfer heat generated in the core.

Differential pressure instrumentation, Heated junction thermocouples (HJTCs) and instrumentation to measure reactor coolant level across the vertical span of the reactor vessel outlet nozzles are provided to give the operator unambiguous indication of RCS water level during reduced inventory operations.

4.10.2.2 Reactor core and fuel design

The reactor core consists of 241 fuel assemblies and 93 or more control element assemblies (CEAs). Each fuel assembly is a 16 x 16 array consisting of 236 fuel and poison rods and 5 guide tubes. The fuel rods are Zircaloy tubes containing slightly enriched uranium dioxide pellets.

Erbia burnable absorber rods having Er_2O_3 admixed in enriched UO_2 in the central rod portion (axially) and natural or low enrichment UO_2 at the top and bottom are employed. The rods are mechanically similar to fuel rods. The total column length is the same as the column length in fuel rods.

The control element assemblies consist of either four or twelve neutron absorber elements arranged to engage the peripheral guide tubes of fuel assemblies. The four-element CEAs are used for control of power distribution and core reactivity in the power operating range. The twelve-element CEAs make up the balance of the control groups and provide the core with strong shutdown rods.

Full-strength CEAs consist of Inconel clad with boron carbide or silver-indium-cadmium absorber rods. Reduced strength control rods composed of solid Inconel provide the capability to change operating power level using control rods only. The System 80+ approach simplifies reactivity control during plant load changes and reduces liquid waste processing requirements that normally accompany changes in soluble boron concentration.

System 80+ includes various features to accommodate 100% mixed oxide cores such as: increased reactor coolant system and spent fuel decay heat removal capacity; ability to accommodate higher boron concentrations; reactor vessel and internals design that accommodates increased fluence and gamma heating; and, features that allow additional control rods to meet shutdown margin requirements.

4.10.2.3 Fuel handling and transfer systems

Fuel handling equipment provides for the safe handling of fuel assemblies and CEAs under all specified conditions and for the required assembly, disassembly, and storage of the reactor vessel head and internals during refuelling.

The major components of the system are the refuelling machine, the CEA change platform, the fuel transfer system, the spent fuel handling machine, and the new fuel and CEA elevators. This equipment is provided to transfer new and spent fuel between the fuel storage facility, the containment building, and the fuel shipping and receiving areas during core loading and refuelling operations. Fuel is inserted and removed from the core using the refuelling machine. During normal operations, irradiated fuel and CEAs are always maintained in a water environment.

4.10.2.4 Primary components

Reactor pressure vessel

The reactor vessel is designed to contain and support the core and nuclear fuel. The design is based on the well proven System 80 design. A major improvement in manufacturing and operation has been achieved through the use of ring forging which reduces the number of welds and the overall complexity of the vessel. The remaining welds have been relocated to areas of lower neutron flux, thus enhancing the

vessel's resistance to brittle fracture. The reactor vessel is designed and constructed in accordance with the ASME code, while the test and inspection requirements of the reactor vessel exceed those of the ASME code.

To minimize the effects of irradiation on material toughness properties of core belt line materials, restrictions on upper limits for residual elements that directly influence the RT_{NDT} (the temperature at which the steel becomes non-ductile) shift are required by the design specification. Specifically, upper limits are placed on copper, nickel, phosphorous, sulfur, and vanadium.

The maximum integrated fast neutron flux exposure of the reactor vessel wall opposite the mid plane of the core is less than 6.2E19 nvt. This value assumes a sixty-year vessel design life and an eighty percent plant capacity factor. The maximum expected increase in transition temperature is about 34°C (93°F). Based on an initial RT_{NDT} of -12°C (10°F), no operating restrictions are necessary to limit vessel stresses.

Reactor internals

Reactor vessel internals are manufactured as all-welded structures. They consist of the core support barrel assembly, which includes the core shroud and the core support structure, and the upper guide structure assembly. The core support barrel assembly provides support and location positioning for the fuel assemblies and contains instrument guide paths and hydraulic flow paths. The upper guide structure assembly provides an insertion path and lateral support for the control element assemblies.

Steam generators

The two steam generators are vertical U-tube economizer heat exchangers. The System 80+ steam generators incorporate several design enhancements including better steam dryers, increased overall heat transfer area, slightly reduced full power steam pressure and better access for maintenance. The design also provides a larger secondary feedwater inventory which extends the "boil dry" time, thereby enhancing the plant's capability to tolerate upset conditions and improving operational flexibility.

The corrosion resistance properties of the System 80+ steam generators are enhanced by the following design features:

- steam generator tubes made of thermally treated Inconel 690;
- steam, feedwater and condensate systems employing materials resistant to corrosion and the generation of corrosion products which can be transported into the steam generators;
- high capacity steam generator blow down system; and
- secondary circulation system for chemistry control during wet lay up.

The System 80+ Standard Plant Design incorporates one N-16 gamma ray monitor per steam generator to provide a sensitive and specific indication for primary coolant leakage through steam generator tubes, in addition to monitors in the main condenser evacuation system and the blowdown system.

Pressurizer

The reactor coolant system (RCS) operates at a nominal pressure of 15.5 MPa (2250 psia). The pressurizer has a 33% increased operating volume relative to previous designs to enhance transient response.

Reactor coolant pumps

The four reactor coolant pumps (RCP) are vertical, single stage, bottom suction, horizontal discharge, motor-driven centrifugal pumps. The pump and motor shafts are directly connected by a low vibration flexible coupling.

The shaft seal assembly consists of two face-type, mechanical seals in series, with controlled leakage bypass to provide the same pressure differential across each seal. The seal assembly is designed for 17.25 MPa (2500 psi) differential pressure and to reduce the leakage pressure from Reactor coolant system

pressure to the volume control tank pressure. A third, face-type, low-pressure vapour seal at the top is designed to withstand system operating pressure when the pumps are not operating.

The System 80+ RCP shaft seals are cooled by (1) seal injection water from the Chemical and volume control system, and (2) the Component cooling water system (CCWS) through a high pressure seal cooler. Pump operation may continue indefinitely provided either seal injection flow or the CCWS is available. The System 80+ design also includes an additional diverse and dedicated seal cooling system, the Dedicated seal injection system (DSIS), which features a positive displacement pump to provide a diverse means of seal injection to the RCPs if normal means of seal cooling are lost. In particular, the DSIS provides enhanced capability to cope with a Station blackout (SBO).

Main coolant lines

The reactor coolant system (RCS) comprise, as noted above, two independent parallel loops in a "two hot legs and four cold legs" arrangement. The hot legs consist of 1065 mm (42 inch) ID pipes, and the cold legs of 760 mm (30 inch) ID pipes.

4.10.2.5 Reactor auxiliary systems

Chemical and volume control system (CVCS

The CVCS controls the purity, volume, and boric acid content of the reactor coolant. The CVCS is not required for any safe shutdown or accident mitigation function. Centrifugal charging pumps are used instead of reciprocating pumps as in previous designs, resulting in a significant increase in reliability.

The design pressure of an appropriate portion of the charging pump suction line has been increased to at least 6.3 MPa (900psig). This higher design pressure significantly reduces the chance of a large interfacing system LOCA, even at full RCS pressure.

Component cooling water system (CCWS)

The CCWS comprises two separate, independent, redundant, closed loop, safety-related divisions. Either division of the CCWS is capable of supporting 100% of the cooling functions required for a safe reactor shutdown.

The CCWS design precludes the need for raw water cooling of components within the Nuclear Island structures. Therefore, there is no potential for flooding within the nuclear island structures from raw water sources (i.e., the station service water system), and the number of components and amount of piping exposed to corrosive water and potential biofouling is minimized. The CCWS operates at a higher pressure than the SSWS, thus preventing the leakage of station service water into the CCWS in the event of a CCW heat exchanger tube leak.

Station service water system (SSWS)

The SSWS consists of two separate, redundant, open loop, safety-related divisions, each cooling one of two divisions of the component cooling water system (CCWS), which in turn cools 100% of the safety-related loads.

Four identical station service water pumps are provided, two pumps per division. Manual start and stop actuation of the station service water pumps is provided from the control room to override automatic actuation. Each pump provides 100% of the required flow for post-LOCA conditions. Typically, during normal operation only one pump per division is operating. The second pump in the respective division will automatically start on a low pump discharge pressure signal.

Pool cooling and purification system (PCPS)

The PCPS consists of the spent fuel pool cooling system and the pool purification system. The PCPS is designed to remove the decay heat generated by the stored spent fuel assemblies from the spent fuel pool water, and purify the contents of the refueling pool during refueling operations. System piping is arranged

so that failure of any one pipeline cannot drain the spent fuel pool below the water level required for radiation shielding.

Liquid waste management system (LWMS)

The LWMS provides a means to collect, store, process, sample, and monitor radioactive liquid waste. The LWMS has appropriate instruments and controls to permit most operations to be conducted remotely.

Radioactive wastes are segregated by routing to an initial collection sump or tank. This permits more effective processing of each type of waste and may lead to reduced solid waste volumes. The principal waste treatment process is ion exchange. Processed liquid radioactive waste is sampled prior to release from the monitor tanks. Radiation monitors are provided in the discharge line.

Gaseous waste management system (GWMS)

The GWMS uses charcoal at ambient temperature to delay the passage of radioactive gases through the system. When operating at design conditions, the mass of charcoal provided in the adsorber beds is sufficient to provide a minimum of 30 days delay for xenon and a minimum of 3 days delay for krypton. The GWMS operates at pressures slightly above atmospheric, thus limiting the potential for oxygen in-leakage. Welded connections are used wherever not restricted due to maintenance requirements. All control valves are provided with bellows seals to minimize leakage through the valve topworks. The GWMS is designed for monitoring and purging of any oxygen concentrations above 1 % in the atmosphere.

Solid waste management system (SWMS)

Primary functions of the SWMS include providing means by which spent resin, filters, etc., from the LWMS and primary letdown systems are processed to ensure economical packaging within regulatory guidelines, as well as handling dry, low activity wastes for shipment to a licensed burial facility. Capability is provided for solidification of dewatered resins or sluicing to containers approved for shipping and disposal of dewatered ion exchange resins. Also, connections are provided for use of vendor supplied services such as rapid dewatering or waste drying systems when it is determined that the use of these methods represents a savings over the permanently installed alternatives. A shielded onsite storage area is provided that is capable of storing the maximum number of full shipping containers generated in any one year period containing the greatest expected waste generation.

Heating, ventilation and air conditioning (HVAC) systems

HVAC systems are provided as required throughout all areas for personal comfort, personnel safety protection, and equipment functional protection. The general flow path in the HVAC systems is from the radiologically clean areas of the plant to the contaminated areas.

For those buildings and rooms required for functional use during all plant operating modes (normal, shutdown, emergency), two separate HVAC systems are provided although in most cases some safety-related air distribution ducting is shared. Redundant Seismic Category I trains are provided for safety-related systems, and radiation monitors are located upstream of filter inlets. Ventilation systems are division-specific so that fire or smoke in an area containing a safety-related division of equipment cannot migrate through the ventilation ducts to an area containing the redundant division of safety-related equipment.

The nuclear annex, fuel building, and reactor building subsphere are maintained under negative pressure with respect to the atmosphere. The leakage taking place from one of these areas to the other is filtered before it is released to the atmosphere.

Fire protection system

The fire pumps are located in the yard area in a non-seismic and non-safety-related structure. The fire pump house is subdivided into two separate fire areas by a three-hour rated fire barrier. The diesel-driven fire pump and its associated controller, fuel tank, piping, and fittings are located in one of the fire areas. The electric motor-driven fire pump, and electric motor-driven pressure maintenance pump (Jockey pump), their

associated controllers, piping and fittings are located in the other fire area. The fire protection water supply is provided by two, 1136 m^3 (300 000 gallon) ground level storage tanks.

Each pump is arranged to take suction from either tank. The diesel fuel oil storage tank is sized to provide an eight-hour fuel supply to the diesel engine driven fire pump. Back-up power for the electric motor driven fire pump is provided by the site alternate AC power supply combustion turbine and is electrically protected so that fire in the power house will not interrupt pump operation. Discharge piping of each pump is interconnected so that either pump can supply either connection to the underground water distribution system.

Two pipes penetrate containment to provide redundant water supplies to the primary and back-up fire protection systems. Fire hose and standpipe systems located in the reactor building and the nuclear annex meet Seismic Category I requirements. The primary water supply to the standpipe system is from the fire protection water distribution system. A 68 m^3 (18 000) gallon Seismic Category I water storage tank supplying a 34 m^3/h (150 gpm) Seismic Category I pump is also connected to the fire hose standpipe system.

4.10.2.6 Operating characteristics

The System 80+ reactor core is designed to operate at a maximum core power level of 3914 MWt. At this core power level, the total thermal output is 3931 MWt. The turbine generator provides a net power of approximately 1350 MWe.

The Turbine bypass system is capable of relieving 55% of full main steam flow to dissipate heat from the reactor coolant system during turbine and/or reactor trip. The turbine bypass system, in conjunction with the reactor power cutback system, can accommodate a 100% load rejection without a reactor or a turbine trip and without lifting either the primary or secondary safety valves.

The System 80+ design can continue to operate at 100% power upon loss of a feedwater or a condensate pump.

The manoeuvring capability of the System 80+ design includes the following:

- Step power changes of ± 10% in the 15%-100% power range
- Power ramps of ± 5% per minute in the 15%-100% power range
- Daily load cycle of 100%-50%-100%

4.10.3 Description of turbine generator plant system

4.10.3.1 Turbine generator plant

The steam and power conversion system converts the heat energy generated in the reactor into electrical energy. A turbine bypass system and atmospheric dump valves are available to dissipate heat from the reactor during a turbine and/or reactor trip. The turbine generator consists of a double-flow, high-pressure turbine and three double-flow low pressure turbines driving a direct-coupled generator. Generator rating, temperature rise, and class of insulation are in accordance with IEEE standards. Differential relays protect the generator against electrical faults.

Mechanical overspeed protection provides one mechanical overspeed trip device whose setpoint is 110% of rated speed. Electronic overspeed protection trips at 112% of rated speed. To further decrease the possibility of an overspeed condition there are two redundant reverse power relays. The turbine safety system is independent of the turbine control system.

The turbine speed controller, including valve position controllers, uses a 1-out-of-2 scheme of redundancy. There is automatic switchover (bumpless) from one controller to the other in case of a disturbance on one controller.

4.10.3.2 Condensate and feedwater systems

The entire condensate system is non-safety-related. The portions of the feedwater system that are required to mitigate the consequences of an accident and allow safe shutdown of the reactor are safety-related. Three 50% capacity motor-driven condensate pumps (two operating and one standby) deliver condensate from the condenser hotwell to side stream full flow condensate polisher cells and three parallel strings of low pressure (LP) feedwater heaters to the deaerator. Three 50% capacity motor-driven feedwater booster pumps (three operating) deliver condensate from the deaerator storage tank to the suction of the main feedwater pumps. Each booster pump shares a common drive and shaft with a main feedwater pump. Adjustable speed motor drives are used. Three 50% capacity motor-driven main feedwater pumps (three operating) deliver feedwater through two stages of two parallel high pressure feedwater heaters to a single feedwater distribution header. Feedwater flow is then directed to each steam generator. The feedwater control system automatically maintains proper steam generator level from 5% to 100% power.

During shutdown and startup, a motor-driven startup feedwater pump provides feedwater from the deaerator storage tank or condensate storage tank. The startup feedwater pump is capable of providing 0-5% of full power feedwater flow to both steam generators in addition to pump recirculation requirements.

4.10.3.3 Auxiliary systems

Essential auxiliary systems for the plant are described in Section 4.10.2.5 above.

4.10.4 Instrumentation and control systems

4.10.4.1 Design concepts including control room

The Nuplex 80+ advanced control complex (ACC) (Figure 4.10-3) is a plant-wide computer-based control and monitoring systems design, of which the overall architecture is shown in Figure 4.10-4. The overall Nuplex 80+ instrumentation and control complex consists of Main control panels (MCPs), Remote shutdown panel (RSP), Discrete indicating and alarm system (DIAS), Data processing system (DPS), and Component control system (CCS). The ACC makes extensive use of remote multiplexing, digital computers, colour graphic displays and fiber-optic data communications. The control complex design integrates monitoring and control of both nuclear and balance-of-plant systems.

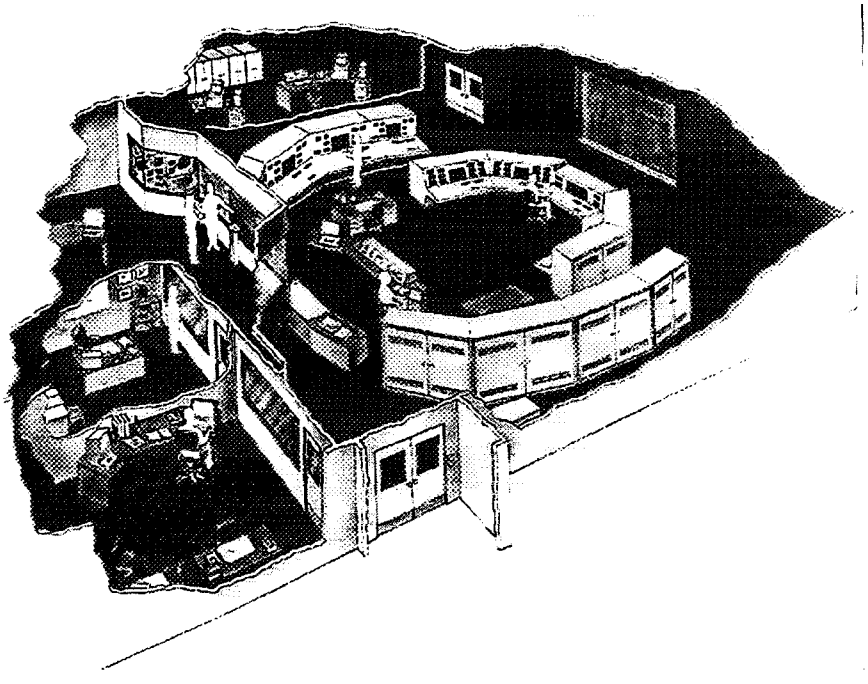


Figure 4.10-3 Nuplex 80+ Advanced Control Center

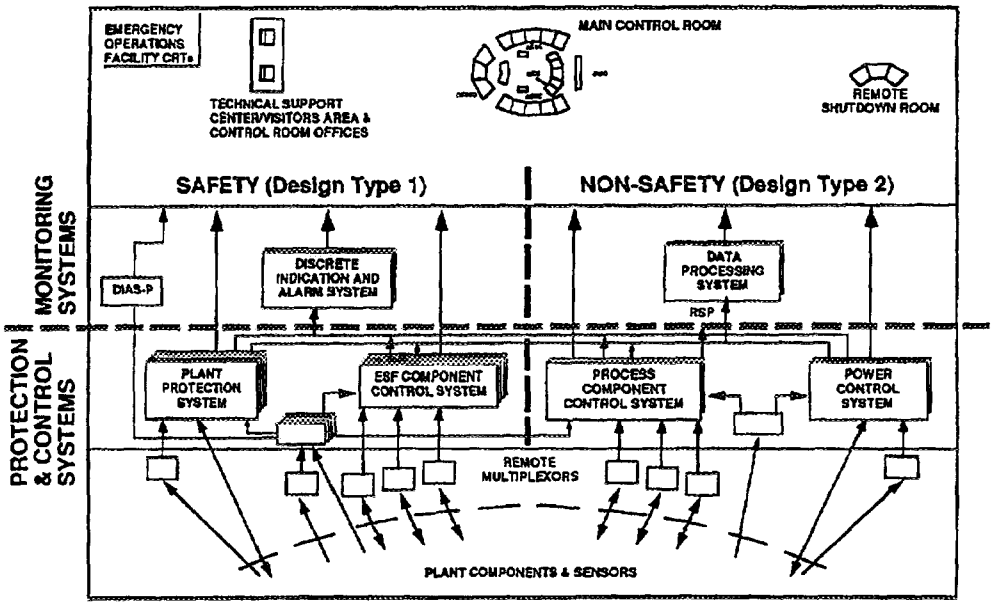


Figure 4.10-4 Overall Nuplex 80+ instrumentation and control architecture

The master control console, consisting of five MCPs, is designed for one person seated operation of the plant from hot standby through full power modes of operations. However, the main control room design accommodates two control room operators and a supervisor for all normal modes of plant operation and additional operating staff during emergencies. The arrangement and layout of the MCPs were established based on the coordinated design effort of a team of human factors specialists, reactor operators, instrumentation and control engineers, architectural engineers and owner utility designers.

The RSP design includes a minimum of two redundant channels of the safety-related instrumentation and controls (isolated from circuits routed to the main control room) necessary to achieve hot standby if the main control room must be evacuated. Additionally, local controls, RSP controls, and instrumentation are provided to bring the plant to cold shutdown conditions using applicable procedures.

The DIAS is a fixed position indication and alarm system that utilizes flat panel display devices. The DIAS allows continuous monitoring of safety functions.

The DPS is a fault tolerant multiprocessor computer based system which provides plant data and status information to the operations staff. The major functions performed by the DPS include plant wide data acquisition via dedicated data links to other plant systems, validation of sensed parameters, execution of application programs and performance calculations, monitoring of general plant status and plant safety status, generation of logs and reports, the determination of alarm conditions, sequence of events recording, and post-trip review.

The CCS is designed to control discrete-state components such as pumps, valves, heaters and fans within plant systems. The CCS consists of the Engineered safety features-Component control system (ESF-CCS) and Process-Component control system (P-CCS) assemblies to provide control for the different channels of Class 1E equipment, as well as non-Class 1E equipment. The Nuplex 80+ ACC includes an integrated plant status overview (IPSO) panel, a large screen display device, that provides the operators and supervisory staff with a quick means of assessing the plant status from anywhere in the controlling work space.

4.10.4.2 Plant protection system and other safety systems

System 80+ uses an all digital I&C design. The Plant protection system (PPS) (Figure 4.10-5) is a safety-related I&C system which consists of the Reactor protection system (RPS) and the Engineered safety features actuation system (ESFAS). Plant control systems are strictly separated from these safety systems, including the respective hardware and software.

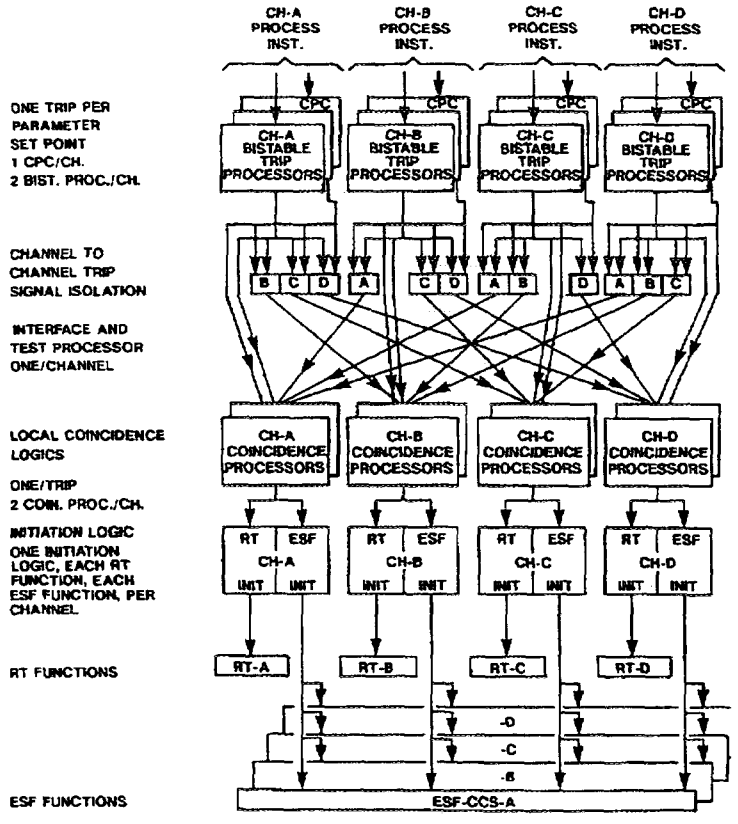


Figure 4.10-5 Plant protection system

The PPS is augmented by the Alternate protection system which generates an alternate reactor trip signal and an alternate emergency feedwater actuation signal that are independent and diverse from the PPS. The PPS employs automatic on-line functional testing to eliminate most periodic surveillance tests.

The RPS is segregated into four completely independent channels. This enhances operability by allowing the RPS to be operated with one channel by-passed while still requiring a coincidence of two channels in a tripped condition for system actuation.

A completely redundant and diverse backup to safety-grade control and monitoring systems is included to resolve regulatory concerns regarding common cause failures of digital protection systems.

4.10.5 Electrical systems

A single line diagram of the electrical distribution systems is shown in Figure 4.10-6.

4.10.5.1 Operational power supply systems

The on-site power system for the unit consists of the main generator, the generator circuit breaker, unit main transformer, the unit auxiliary transformers, reserve auxiliary transformers, the diesel generators, an alternate AC source, the batteries, and the auxiliary power system. The two off-site power circuits (to the switchyards) are designed to be independent and physically separate to assure their availability under normal and postulated accident conditions.

Each of the two reserve auxiliary transformers is sized with power capability to supply the most conservative power requirements of its associated Class 1E buses, the most conservative power requirements of its associated permanent non-safety bus, and the power requirements for at least one reactor coolant pump and its support loads. This rating is the self-cooled rating of the transformer. Additional margin allows future load growth.

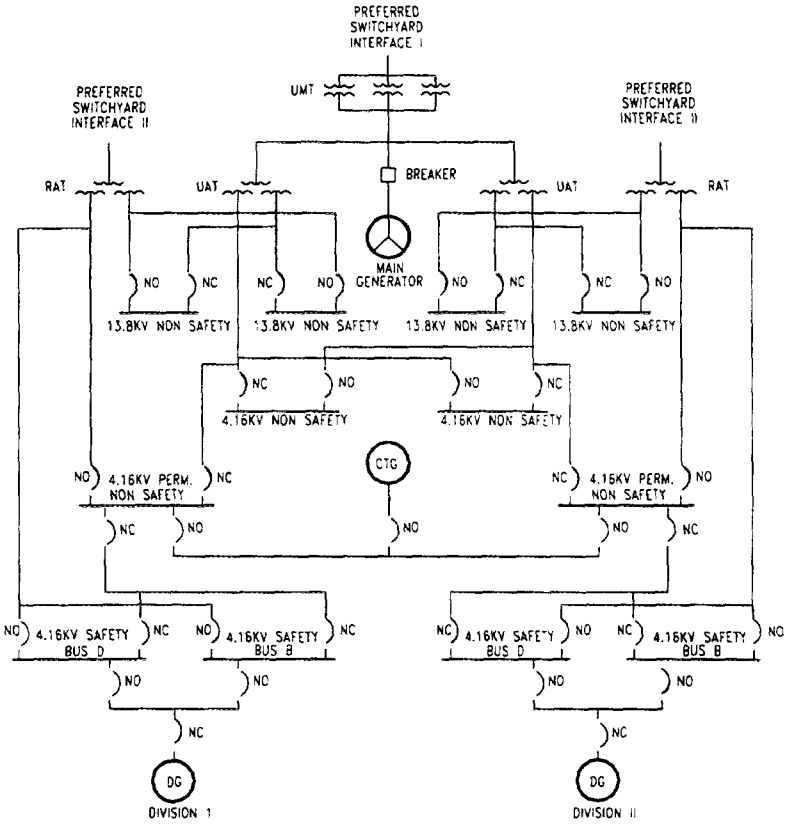


Figure 4.10-6 Electrical Distribution System

The two unit auxiliary transformers are sized to provide the full load requirements of the buses in their load group within their self-cooled rating. Again, additional margin allows future load growth.

The unit also has a 125V DC Auxiliary control power system and a 250V DC Auxiliary power system to supply essential Non-Class 1E DC loads. Additionally, this system also provides power to Non-Class 1E 208/120V AC loads through inverters.

4.10.5.2 Safety-related electrical systems

The Class 1E safety loads are divided into two redundant and independent load groups, Divisions I and II. If both the off-site power sources and the standby emergency diesel generators are unavailable, either one of the divisions may be powered independently from the Alternate AC (AAC) source.

A 125V DC Vital instrumentation and control power system provides power to the Class 1E DC loads and the diesel generators. Additionally, this system provides power to Class 1E 120V AC loads through inverters.

The vital batteries have adequate capacity, without chargers, to provide the necessary DC power to perform the required safety functions in the event of a postulated accident assuming a single failure. In particular, battery capacity is sufficient to provide an eight hour coping time for station blackout.

4.10.6 Safety concept

4.10.6.1 Safety requirements and design philosophy

The System 80+ design improves upon the existing high level of safety achieved in its predecessor, the System 80 design, by incorporating enhancements based on Probabilistic Safety Assessment (PSA) insights, guidance from the EPRI ALWR Utility Requirements Document (URD), and the NRC's Severe Accident Policy.

Deterministic design basis

The System 80+ Standard Plant Design has been analyzed in accordance with NRC Standard Review Plan requirements to ensure that it can withstand anticipated operational occurrences as well as a broad spectrum of postulated accidents without posing undue risk to the public health and safety.

The System 80+ design meets all NRC requirements for design basis events, and does so with considerable margin. For example, the LOCA 2-hour thyroid dose limit at the site boundary is 3 Sv (300 rem). The design basis LOCA analysis for System 80+ showed the calculated thyroid dose to be 1.72 Sv (172 rem). A realistic LOCA analysis resulted in a calculated thyriod dose of .017 Sv (1.7 rem).

Risk reduction

The System 80+ PSA showed that the estimated core damage frequency (CDF), including consideration of shutdown initial conditions and conservative estimates for fire, flood, and external events, is 2.7E-06 events per year (excludes seismic which is discussed in the following section). This represents a risk reduction of more than two orders of magnitude for System 80+ relative to its predecessor and is well within the goal of 1.0E-05 events per year. Considering only events initiated with the reactor initially shutdown, enhancements to the System 80+ design have reduced the the risk of core damage by a factor of 40.

The large release goal specified for System 80+ is that releases in excess of 0.25 Sv (25 rem) at 0.8 km (0.5 mile) from the reactor shall have an exceedance frequency of less than 1.0E-06 events per year. The System 80+ PSA shows that the probability of having such a release is 5.3E-08 events per year. The probability of having a release in excess of 0.25 Sv (25 rem) at 300 meters is 6.2E-08 events per year, demonstrating that System 80+ can meet the large release goal for smaller sites.

The significantly improved technology reflected in the NRC's new radiological source term, described in report NUREG-1465, has been applied to the System 80+ design. One area of significant benefit, calculated containment spray effectiveness, is illustrated in Figure 4.10-7. Analyses demonstrated that the best-estimate site boundary dose for a severe accident is less than the US Environmental Protection Agency's PAG which are specified at a 16 km (10 mile) radius from the plant. For example, it has been found from the PAG analysis that core melt scenarios would not exceed the PAG 24 hour dose limits at the site boundary [0.8 km (0.5 mile)]. The System 80+ dose for the worst scenario resulted in a site boundary dose of 0.003 Sv (0.3 rem) (Total Effective Dose Equivalent) vs. a PAG of 0.01 Sv (1 rem). Similarly, the thyroid dose is only 0.027 Sv (2.7 rem) vs. a limit of 0.05 Sv (5 rem). This analysis has demonstrated the technical feasibility of simplified emergency planning for the System 80+ design.

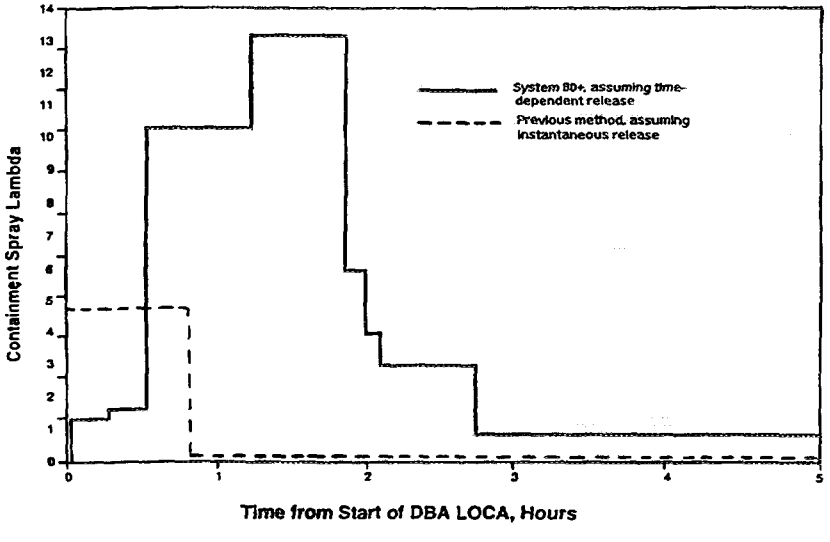


Figure 4.10-7 LOCA - Containment spray effectiveness for particulate cleanup

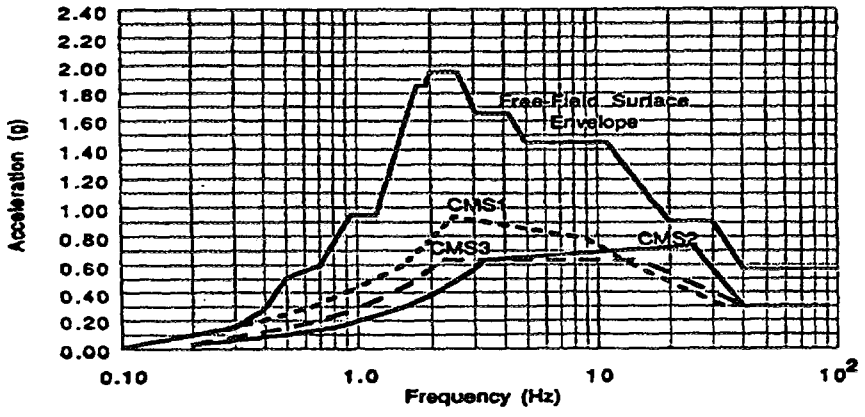


Figure 4.10-8 Envelope of free field ground surface spectra

External and internal hazards

The ability of the System 80+ design to accommodate internal and external hazards is demonstrated by the deterministic design basis analysis and PSA results described above. In accordance with the practice adopted by the NRC, however, the System 80+ seismic capability was demonstrated by a seismic margins assessment, rather than by the PSA

The System 80+ plant has been designed to provide a standard plant which envelopes the seismic requirements for the majority of potential sites in the world without need for further suitability assessments. Development of such a "seismic envelope" involved evaluations of different rock and soil conditions, analysis of structural designs, and the seismic capacity or "fragility" of certain components.

Three control motions anchored to a 0.3g horizontal peak ground acceleration were developed and applied to twelve generic soil sites as well as rock site conditions. The resulting envelope of site free field spectrum (Figure 4.10-8) is essentially equivalent to an NRC Regulatory Guide spectral shape anchored to a maximum ground acceleration of 0.55g. For System 80+, the minimum seismic fragility was determined to be 0.73g. That is, the likelihood of core damage is less than 5% for a 0.73g peak-ground-acceleration earthquake.

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

Safety systems configuration

The System 80+ plant is designed on the basis of two divisions of safety-related equipment, each of which is fully capable of achieving and maintaining a safe shutdown despite the complete loss of the redundant division's functionality. Outside of containment, the two redundant divisions of safety-related equipment are separated by an interdivisional barrier except for the control room and the remote shutdown room which are physically separated and electrically isolated from each other. The interdivisional barrier is a three hour fire rated barrier and a flood barrier. Heating, ventilation, and air conditioning and cooling water systems are designed to maintain divisional separation.

In areas inside the containment where equipment and cabling associated with safe shutdown equipment for the two divisions converge, an alternate capability is provided to achieve and maintain safe shutdown using systems and equipment spatially and electrically isolated from the convergent systems.

Containment isolation system (CIS)

The CIS provides a means of isolating fluid systems that pass through containment penetrations so that any radioactivity that may be released into the containment following a design basis accident will be confined within the steel containment building. The CIS provides a pressure barrier at each containment penetration. Valves that must be isolated are installed with air-operated controllers or motor-operated

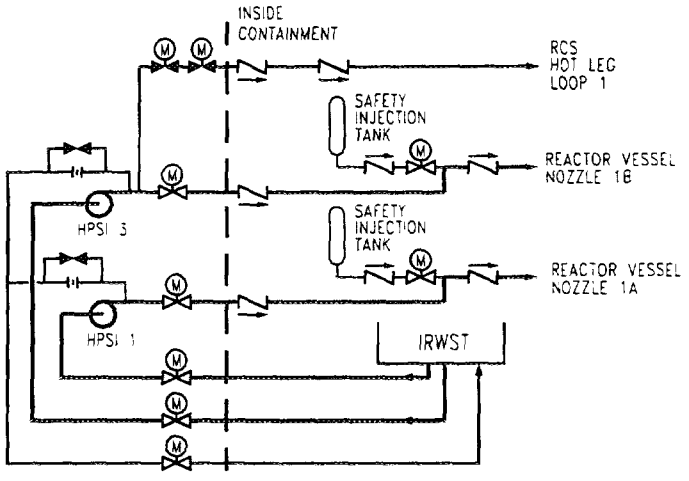


Figure 4.10-9 Safety injection system

controllers. Lines that must remain in service following an accident have at least one remotely-operated, manually controlled valve.

Main steam isolation system (MSIS)

The MSIS isolates the steam line piping and the main feedwater piping associated with a steam generator following a steam generator tube rupture, a main steam line break, or a main feedwater system upset.

Safety injection system (SIS)

The safety injection system (SIS) (Figure 4.10-9) incorporates a four-train safety injection configuration and an In-containment refuelling water storage tank (IRWST).

The SIS utilizes four 100% high pressure safety injection pumps to inject borated water directly into the reactor vessel. In addition, four safety injection tanks are provided. The SI pumps can be tested at design flow during power operation, are normally aligned to the IRWST, and a realignment for recirculation following a LOCA is not required. Severe accident mitigation capabilities of the SIS are discussed in Section 4.10.6.3.

Pressure retaining components outside containment have a design pressure of at least 6.3 MPa (900 psig). This increased design pressure significantly reduces the chance of a large interfacing LOCA, even at full RCS pressure.

Safety depressurization system (SDS)

As a backup to the normal pressure control system and the reactor coolant gas vent system, the safety depressurization system (SDS) (Figure 4.10-10) provides a safety-grade means of depressurizing the RCS. Severe accident mitigation capabilities of the SDS are discussed in Section 4.10.6.3.

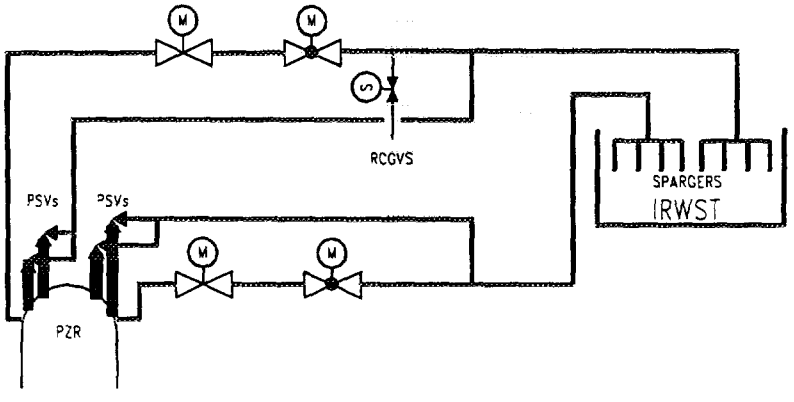


Figure 4.10-10 Safety depressurization system

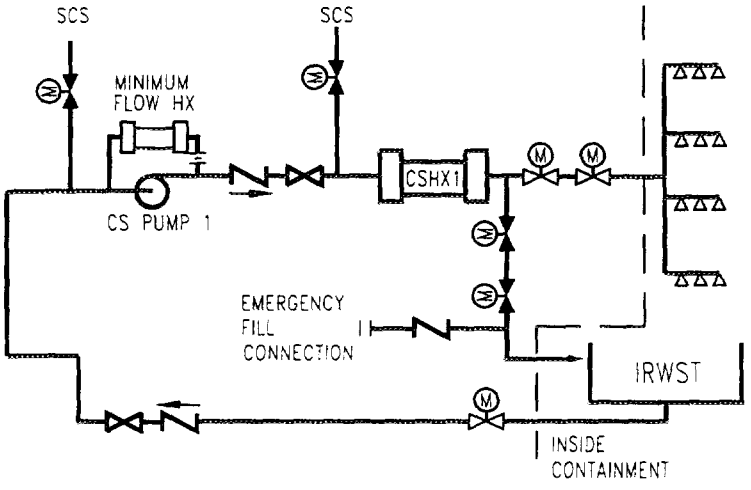


Figure 4.10-11 Containment Spray System

Containment spray system (CSS)

The containment spray system CSS) is designed to maintain containment pressure and temperature within design limits in the unlikely event of design basis mass-energy releases to the containment atmosphere The CSS (Figure 4.10-11) is a fully redundant two-train system. The In-containment refuelling water storage tank (IRWST) is used as the water source for the system. The Containment spray pumps can be manually aligned and used as residual heat removal pumps during SCS operation. Likewise, the SCS pumps can be manually aligned to perform the containment spray function.

The Containment spray system also provides a containment air cleanup function to reduce the concentration of fission products in the containment atmosphere after an accident. No spray additives are required. The emergency fill connection enables water from an external source to be provided to the containment spray headers if the normal spray function is unavailable. Severe accident mitigation capabilities of the CSS are discussed in Section 4.10.6.3. Pressure retaining components outside containment have a design pressure of at least 6.3 MPa (900 psig). This increased design pressure significantly reduces the chance of a large interfacing LOCA, even at full RCS pressure.

Emergency feedwater system (EFWS)

The EFWS (Figure 4.10-12) is a dedicated four-train safety system (two trains per Division) that supplies feedwater to the steam generators for the removal of heat from the RCS in the event the main feedwater system is unavailable following a transient or accident. The EFWS has no operating functions for normal operation. The EFWS is configured into two separate mechanical divisions. Each division is aligned to feed its respective steam generator. Each division consists of one Emergency feedwater storage tank (EFWST), one 100% capacity motor-driven pump subdivision, one 100% capacity steam-driven pump subdivision, valves, one cavitating venturi, and specified instrumentation.

Each common EFW header contains a cavitating venturi. The cavitating venturi restricts the magnitude of the two pump flow as well as the magnitude of individual pump runout flow to the steam generator, and eliminates the need for complicated isolation systems.

A cross-connection is provided between each EFWST so that either tank can supply either division of EFW. A line connected to a non-safety source of condensate is also provided that can be manually aligned for gravity feed to either of the EFWSTs, should the EFWSTs reach low level before Shutdown cooling system entry conditions are reached. Pump discharge crossover piping is provided to enhance system versatility during long-term emergency modes, such that a single pump can feed both steam generators.

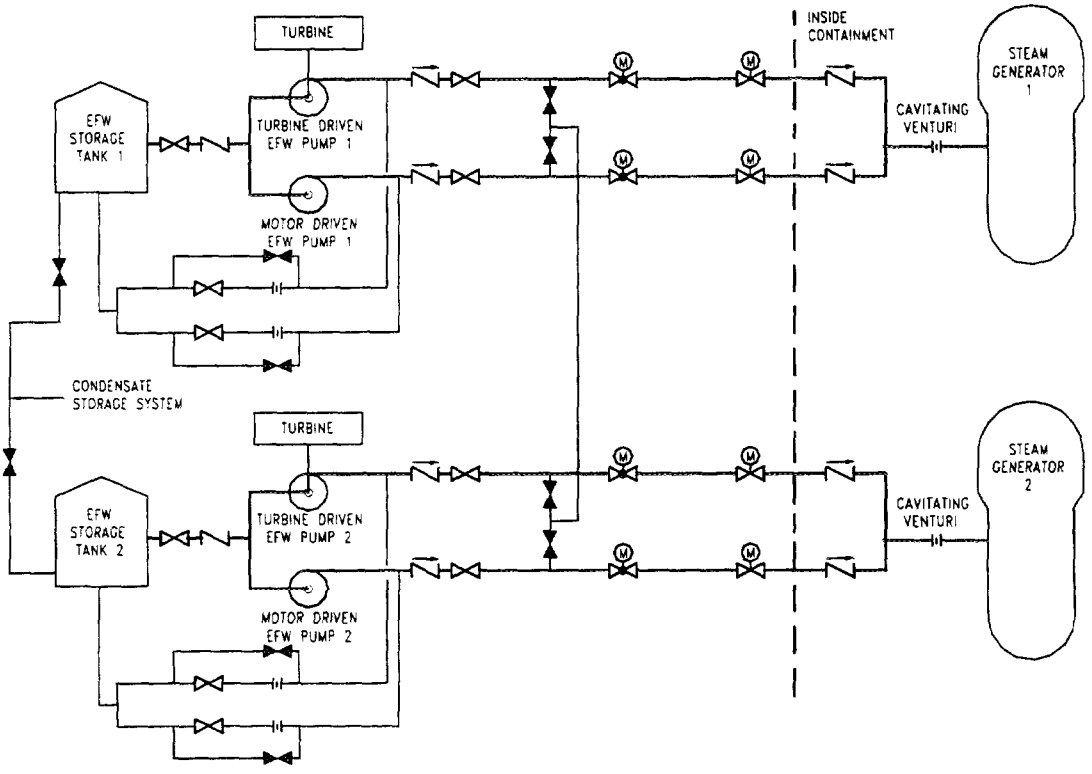


Figure 4.10-12 Emergency feedwater system

Shutdown cooling system (SCS)

The shutdown cooling system (SCS) (Figure 4.10-13) is used to reduce the temperature of the reactor coolant, at a controlled rate, from 176.7° C (350° F) to a refuelling temperature of 48.9° C (120° F) and to maintain the proper reactor coolant temperature during refuelling. Additionally, the SCS is used in conjunction with the atmospheric dump valves (ADVs) and the Emergency feedwater system to cooldown the RCS following a small break LOCA. The SCS is also used subsequent to steam and feedwater line breaks, steam generator tube ruptures, and is used during plant startup prior to RCP restart to maintain flow through the core.

After an accident, the SCS can be put into operation when the RCS pressure and temperature are below approx-imately 2.8 MPa (400 psia) and 193.3°C (380°F). Severe accident mitigation capabilities of the SCS is discussed in Section 4.10.6.3.

The system has a design pressure of 6.3 MPa (900 psig). This higher system pressure provides for greater operational flexibility and significantly reduces the chance of a large interfacing system LOCA, even at full RCS pressure. The SCS pumps do not share functions with the SIS.

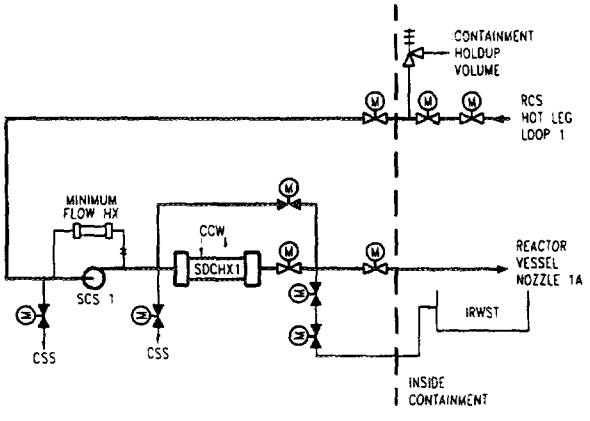


Figure 4.10-13 Shutdown cooling system

Inherent safety characteristics

In the power operating range, the net effect of the prompt inherent nuclear feedback characteristics (fuel temperature coefficient, moderator temperature coefficient, moderator void coefficient, and moderator pressure coefficient) compensates for a rapid increase in reactivity. The System 80+ core is designed to be stable against radial and aximuthal power oscillations. Instrumentation and control capability are provided to readily detect and control axial power oscillations. Despite the stability of the System 80+ core to power oscillations, the power distributions are monitored, and alarms and protection system limits are established to limit the magnitude of asymmetric power distributions should they arise

With respect to flow instability, the available literature, experimental evidence, and core flow stability analyses show that flow instabilities will not adversely affect core thermal margin during normal operation or anticipated operational occurrences.

4.10.6.3 Severe accidents (Beyond design basis accidents)

The System 80+ Standard Plant Design is a more robust plant design that reduces the potential for core damage and moderates the severity of such an accident should one occur. Various features of the System 80+ design are discussed below specifically with respect to severe accident prevention and mitigation.

Safety depressurization system (SDS)

In the context of severe accident prevention, the SDS performs the following functions:

- The reactor coolant gas vent (RCGV) function of the SDS provides a safety-grade means of venting non-condensable gases from the pressurizer and the reactor vessel upper head to the Reactor drain tank (RDT) during post-accident conditions. In addition, the RCGV provides a safety-grade means to depressurize the RCS in the event that pressurizer Main spray and auxiliary spray systems are unavailable.
- The rapid depressurization (RD) function, or bleed function, provides a manual means of quickly depressurizing the RCS when normal and emergency feedwater are unavailable to remove core decay heat through the steam generators.

In the event a high pressure meltdown scenario develops and the feed portion of feed and bleed cannot be established, the SDS can be used to depressurize the RCS to ensure that a High pressure melt ejection (HPME) event does not occur, thereby minimizing the potential for direct containment heating (DCH).

In-containment refuelling water storage tank (IRWST)

Sufficient borated water is stored in the IRWST to meet all post-accident safety injection pump and containment spray pump operation requirements. The IRWST eliminates the need for switching over from injection mode to recirculation mode during emergency core cooling operations and, therefore, eliminates failures associated with the switch-over in existing nuclear power plants. In addition, the IRWST is the source of borated water for cavity flooding at the onset of a severe accident.

Two emergency diesels and standby combustion turbine

Each of the two divisions of class 1E AC power is supplied with emergency standby power from an emergency diesel generator (DG), each provided with a dedicated 125 V DC battery. In addition to the two emergency DGs, the System 80+ design has an alternate standby onsite AC power source. This is a non-safety combustion turbine power source provided to cope with station blackout scenarios. The alternate power source is independent and diverse from the DGs.

Larger pressurizer

The larger pressurizer volume in System 80+, as compared to the existing generation of nuclear power plants, makes the plant response to transients slower with more limited pressure excursions. The larger volume also helps prevent emptying the pressurizer, uncovering the pressurizer heaters, opening the safety valves, and challenging the safety valves with two phase flow.

Larger secondary inventory in the steam generators

Together with an increased downcomer volume, the 25% increase in steam generator inventory helps reduce pressure and temperature fluctuations during transients and increases the time to steam generator dry-out. The time required to deplete the secondary inventory of the steam generators is about 50% longer for System 80+ than for System 80.

Safety injection system (SIS) with direct vessel injection (DVI)

In conjunction with the rapid depressurization system the SIS can be used for "feed and bleed" operation, as an alternate decay heat removal method. For continuous, long-term, post-LOCA (large) cooling of the reactor core, the SIS pumps are realigned to provide simultaneous hot-leg and direct vessel injection to prevent boron crystallization.

The major characteristics of the System 80+ SIS help reduce SIS unavailability to levels below those for the existing generations of nuclear power plants.

Shutdown cooling system (SCS)/containment spray system (CSS)

In addition to their normal, long-term decay heat removal function, the SCS pumps are designed to perform a backup residual heat removal function and cooling of the in-containment refuelling water storage tank (IRWST). In the backup residual heat removal mode of operation, the SCS is used (in conjunction with the rapid depressurization system) as a backup to the safety injection system (SIS) to inject borated water into the reactor core. The CSS pumps can be used to backup the SCS pumps for improved decay heat removal capabilities. The SCS pumps can also be used as backups to the CSS pumps to perform IRWST cooling during "feed and bleed" operations (in beyond design basis events). To further increase the reliability of the containment spray function, the containment spray headers are designed to accept spray flow from an external source of water via a "tee" connection to the spray line.

Containment vessel

The design basis pressure for the containment is 365 kPa (53 psig). The analyses documented in CESSAR-DC demonstrate that pressures resulting from large break LOCAs or main steam line breaks within the containment will not exceed this design pressure. Calculations also indicate that pressure limits determined in accordance with ASME Service Level C criteria range from 895 kPa (130 psig) at an average steel shell temperature of 143°C (290°F) to 826 kPa (120 psig) at a temperature of 232°C (450°)F. The median ultimate failure pressure ranges from 1297 kPa (188 psia) at 66°C (150°F) to 1103 kPa (160 psia) at 232°C (450°F).

Secondary containment

The secondary containment consists of the containment shield building and the annulus between the steel containment vessel and the shield building. The containment shield building, which houses the containment vessel and safety-related equipment, is designed to provide biological shielding and external missile protection for the containment vessel and safety-related equipment. In addition, the Annulus ventilation system (AVS) provides a mechanism for substantially reducing or eliminating unfiltered fission product releases following design basis and severe accidents.

Hydrogen mitigation system (HMS)

The large System 80+ containment is designed to prevent hydrogen buildup by natural circulation and can passively accommodate a metal-water reaction of up to 75% of the core metal without exceeding a hydrogen concentration of 13% by volume. The HMS is designed to accommodate the hydrogen production from 100% fuel clad metal-water reaction and maintain the average containment hydrogen concentration below the 10% limit for a degraded core accident. The HMS includes hydrogen recombiners and two redundant groups of igniters, each group having independent and separate control, and igniter locations to ensure adequate coverage within the containment.

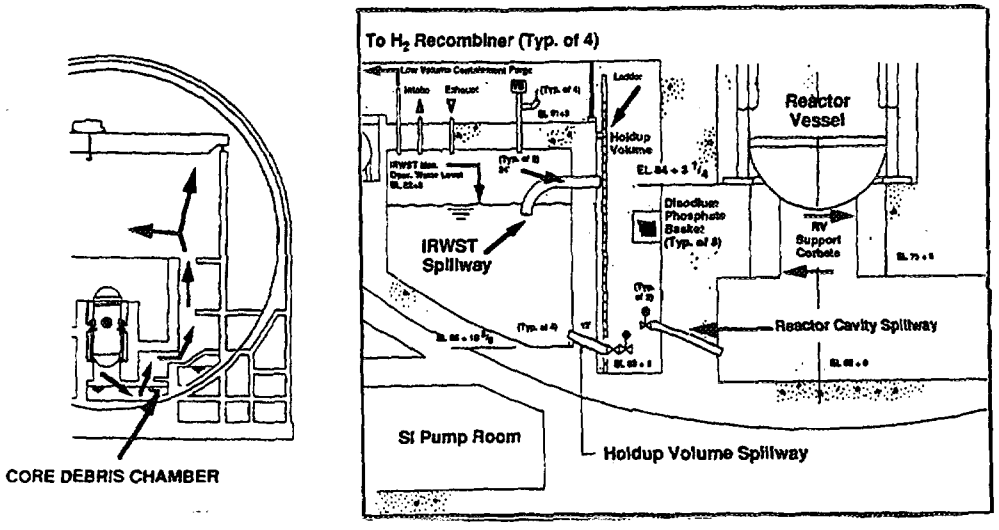


Figure 4.10-14 Reactor cavity and Cavity flooding system

Cavity flooding system (CFS)

The cavity flooding system (CFS) (Figure 4.10-14 on next page) provides a means of flooding the reactor cavity in the event of a severe accident for the purpose of cooling the core debris in the reactor cavity and scrubbing fission product releases.

The cavity flooding system is designed (in conjunction with the containment spray system) to provide an inexhaustible, continuous supply of water to quench the core debris. The CFS is a manually actuated severe accident mitigation system. The CFS takes water from the IRWST and directs it to the reactor cavity. The water flows first into the HVT by way of four 30.5 cm (12 in) diameter HVT spillways and then into the reactor cavity by way of two 25.4 cm (10 in) diameter reactor cavity spillways.

Reactor cavity design

The System 80+ reactor cavity is configured to promote retention of, and heat removal from, the core debris during a postulated severe accident, virtually eliminating the potential for significant DCH-induced containment loadings.

The important features of the System 80+ cavity include:

- A large cavity volume.
- A closed vertical instrument shaft.
- A convoluted gas vent.
- A large recessed corium debris chamber.
- A large cavity floor area.

The cavity includes 906 m^3 (32 000 ft^3) of free volume. Large (and well vented) volumes such as those in System 80+ are not prone to significant pressurization resulting from vessel breach or the corium quench processes.

The vertical instrument shaft permits only limited gas venting, minimizing the potential for corium entrainment and discharge via this path. Corium not entering the shaft will be captured in the large debris retention chamber.

The design of the System 80+ reactor cavity ensures that actual venting to the upper containment either by the vertical instrument shaft or around the reactor vessel flange is small. Thus, steam exits the reactor cavity via a convoluted pathway that includes a 90 degree turn above the top of the core debris chamber and through louvered vents under the refueling pool. As a consequence, the dominant hot gas and potential corium carryover pathway will be to the lower portion of the containment where the containment shell is fully protected by the crane wall. In the highly unlikely event that the corium debris is projected

upward out of the cavity annulus, the Head area cable tray system (HACTS) above the reactor vessel serves to protect the containment shell from a direct corium attack.

The core debris chamber has been sized to hold twice the post-severe accident maximum corium volume. Once deposited in the debris chamber, the debris would be difficult to re-entrain since the retention zone would exhibit a low velocity recirculation flow pattern. Any corium negotiating the 90 degree turn above the debris chamber would be de-entrained by the reactor cavity concrete ceilings and seal table structure.

The reactor cavity is sized and configured to spread out the ejected core debris over the floor surface area during a postulated severe accident so as to meet the EPRI URD criterion of 0.02 m^2 per MWt of surface area below the vessel. The cavity floor and sump are constructed with a limestone aggregate concrete with a minimum $CaCO_3$ content of 17 percent. The flat floor area is free from obstructions to corium debris spreading. The minimum flat floor area for the reactor cavity is 64.4 m^2 (693 ft^2). Within the reactor cavity, the containment shell is protected from corium debris by a concrete basemat layer varying from 0.9 m (3 ft) to about 1.5 m (5 ft) thick. Assuming flooded cavity conditions, it is conservatively estimated that this thickness of concrete flooring will protect the containment shell from corium debris contact for more than 30 hours at its thinnest point. In addition, the structural concrete below the containment pressure boundary provides at least 5 m (16 ft) of concrete which acts to prevent basemat melt-through for at least eight days.

The reactor cavity design differential pressure loading is 1297 kPa (188 psi) with a calculated ultimate differential pressure loading of 1621 kPa (235 psi). This cavity strength is typical of later designed PWR cavities and is robust when compared to the cavity loading expected during a severe accident.

Calculations show that the reactor vessel and the upper cavity could continue to be supported even if the entire lower cavity walls below the corbels were either eroded by corium attack or destroyed by a steam explosion. Alternate calculations have been performed which suggest that a damaged reactor vessel can be supported by the four cold legs.

An ex-vessel steam explosion (EVSE) may occur when corium debris contacts a water pool. EVSE consequences with respect to containment integrity, however, are expected to be insignificant. EVSEs are not expected to be capable of damaging the reactor cavity structures required for support of the reactor vessel or RCS. All "in-cavity" structures that may be damaged by such explosions will be confined to non-load bearing structures and thus will not compromise containment integrity.

Missile protection

During severe accidents, missile protection of the containment shell is primarily accomplished by the use of protective shields and barriers either near the source of the potential missile or in front of the containment shell (such as the crane wall).

Missiles generated via failure of the top head and top head components will be intercepted by the HACTS located directly above the reactor vessel upper head. Thus, consequential damage of the containment steel shell due to failure of a pressurized reactor vessel is highly unlikely.

Improved normally operating component cooling water system (CCWS)

The component cooling water system (CCWS) is a closed-loop system that provides cooling water to remove heat from plant systems, components, and structures. Each division contains two pumps: one is normally operating, while the other pump is in standby and starts automatically if the operating pump trips. This configuration eliminates the demand failures of pumps and valves that were found to be significant contributors to risk.

Emergency feedwater system (EFWS)

The EFWS consists of two divisions, each of which is aligned to deliver feedwater to its respective steam generator. Each division contains a motor-driven pump and a turbine-driven pump. For station

blackout sequences, the turbine-driven pumps of the EFWS are available to remove decay heat from the RCS. Because of the redundancy and diversity of the emergency feedwater trains, this system is a significant contributor to risk reduction.

Separate startup and emergency feedwater systems

The use of a non-safety-related startup feedwater system (SFWS) for normal startup and shutdown operations helps reduce the demands on the emergency feedwater system (EFWS). In addition, the SFWS provides an independent means of supplying feedwater to the steam generators for removing heat from the reactor coolant system (RCS) during emergency conditions when main feedwater is not available (the SFWS is automatically actuated upon loss of main feedwater and prevents the need to actuate the EFWS).

Improved control room design

The System 80+ advanced control room design (Nuplex 80+) improves upon existing control rooms while maintaining their strengths. It is an evolutionary design that provides more and better information to the operator, with corresponding improvements in operator performance. Improved features include prioritized alarms, parameter processing and validation, mode-dependent alarms, integrated normal/post-accident instrumentation, and hardwired backups to digital safety instrumentation and controls.

Facilities design

Facilities are designed to provide physical separation of systems or trains of systems that perform redundant safety-related functions. This increases the availability of systems due to their protection from failures associated with internal fires, internal floods, sabotage, and similar common-cause failures. This contributes to risk reduction when compared to existing plant designs.

Multiple independent connections to grid and turbine/generator runback capability

The System 80+ design includes a main switchyard for incoming and outgoing electric power and a separate and independent backup switchyard (with two safety-grade reserve transformers) that is tied to the grid at some distance from the main switchyard. In addition, the System 80+ turbine generator system and associated buses are designed to runback to maintain house load on a loss of grid event. These features, in conjunction with the combustion turbine generator, reduce the frequency of Loss of Off-site Power (LOOP) events and station blackout events.

Vital batteries

Six independent and separate 125 V DC batteries are included in the System 80+ design, in comparison to four batteries for the System 80 design. For System 80+, each battery can supply the continuous emergency load of its own load group for a period of 2 hours. In addition, the batteries provide a station blackout coping capability assuming manual load shedding or the use of a load management program. This permits operation of the instrumentation and control loads associated with the turbine-driven emergency feedwater pumps for a minimum of 8 hours.

4.10.7 Plant layout

4.10.7.1 Buildings and structures, including plot plan

The Nuclear island structures consist of the Reactor building and the Nuclear annex. Main other buildings are the turbine building and the radwaste building. A plot plan indicating the relative locations of structures within the System 80+ scope is shown in Figure 4.10-15 on next page.

Selection of the basic layout of the System 80+ design included evaluation of factors such as access, laydown area, and separation of redundant equipment. Among the various containment concepts within which the System 80+ design could be implemented, a spherical containment was selected because of advantages discussed in various subsections below. The System 80+ design is not based on a specific site. It envelopes the design basis earthquakes at the majority of potential plant sites in the continental USA. A safe shutdown earthquake (SSE) peak ground acceleration of 0.3 g has been selected. The great seismic capability of the System 80+ design is discussed in Section 4.10.6.1.

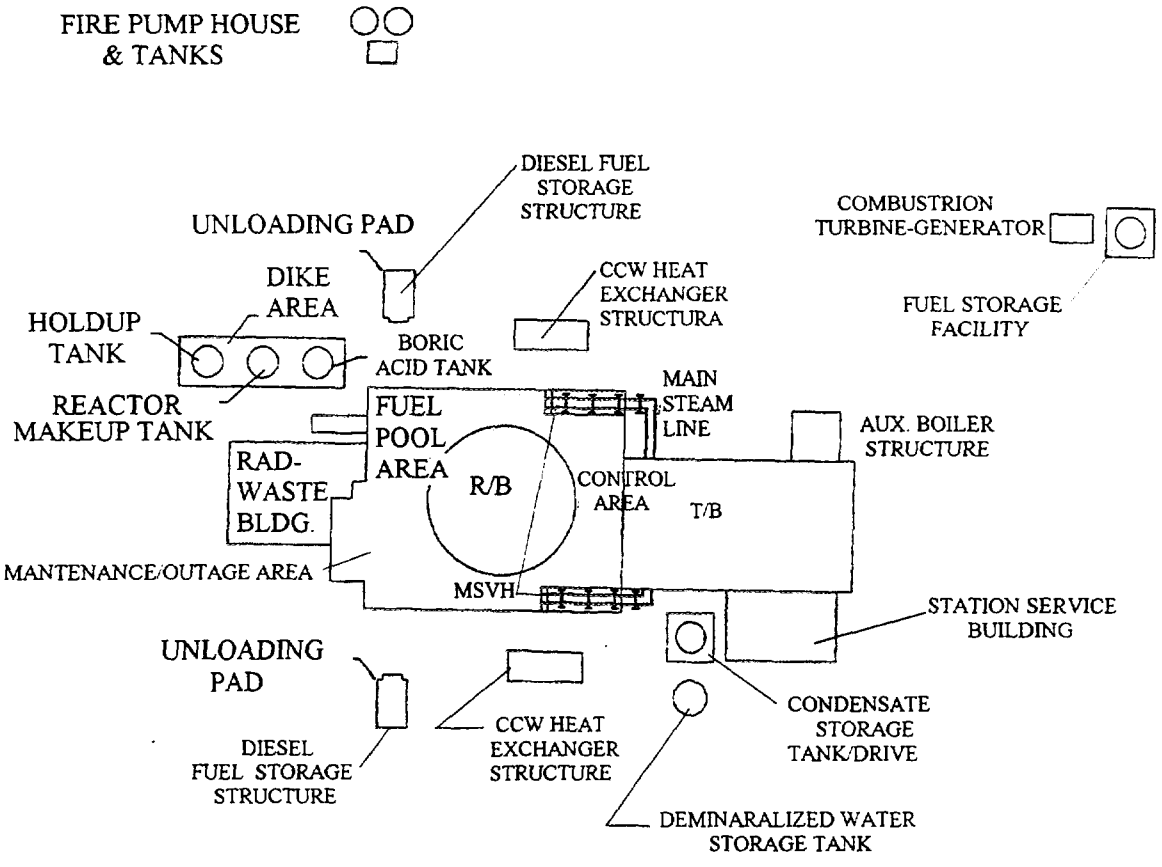


Figure 4.10-15 Plot plan

The concrete shield building is designed to withstand, without loss of function, the loads and effects of any one of the most severe natural phenomena (wind, hurricane and tornado). A design wind velocity of 49 m/s (110 mph) and a maximum tornado wind velocity of 147 m/s (330 mph) are used.

With respect to external industrial hazards and airplane crash, it may be noted that frequent external hazards are treated deterministically based on criteria given in US. NRC Regulatory Guides (administrative control of transportation and storage of hazardous materials on-site, toxic gas and smoke monitors closing control room ventilation intake, site selected outside the radius of influence of potential off site hazardous materials), and infrequent external hazards are evaluated in the PSA.

The potential frequency of aircraft hazards is minimized by siting criteria. Using the siting criteria (plant to airport distance, plant to edge of military training route, plant to edge of federal air-way holding pattern or airport), the probability of an aircraft impact at the site which leads to core damage is less than 10^{-8} events per year. Additionally, the System 80+ design features a dual protection approach. The inner, leak-tight sphere of welded steel is surrounded by the reinforced concrete shield building. This secondary containment protects the internal steel containment from external hazards. Redundancy in the electrical distribution system with physical separation protects against a single transportation accident causing a loss of offsite power (LOOP). The ultimate heat sink has redundancy to ensure a single transportation accident can not cause loss of heat sink.

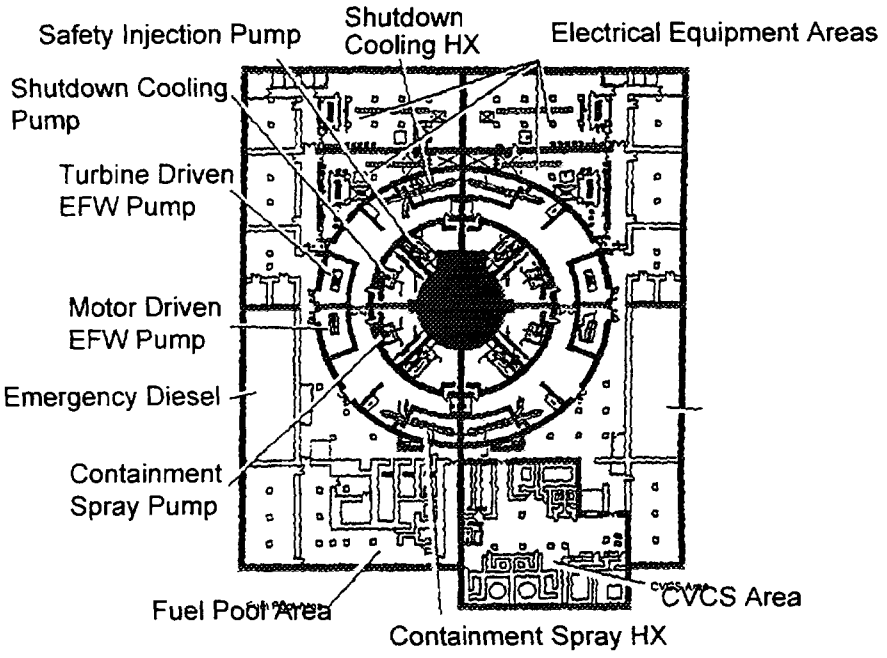


Figure 4.10-16 General arrangement of containment and nuclear annex (basemat level)

4.10.7.2 Reactor building

The reactor building is composed of the containment shield building, spherical steel containment vessel including the internal structures, and subsphere (Figures 4.10-16 and 4.10-17). Structures are Category I except platforms that do not support Seismic Category I equipment and miscellaneous steel; these structures are Seismic Category II. Seismic Category II structures are designed for the SSE using Seismic Category I criteria.

The containment shield building is a reinforced concrete structure composed of a right cylinder with a hemispherical dome. The containment shield building shares a common foundation base mat with the Nuclear annex. The containment shield building houses the steel containment vessel and safety-related equipment located in the subsphere, and is designed to provide biological shielding as well as external missile protection for the steel containment shell and safety-related equipment.

The containment shield building has an inner radius of 64 m (105 ft) and a cylinder thickness of 122 cm (4 ft) up to elevation 16.8 m (55 ft) above finished grade level. Above that elevation, the shield building thickness is 90 cm (3 ft) including the dome area.

The height of the containment shield building is approximately 65.5 m (215 ft). A 1.5 m (5 ft) annular space is provided between the containment vessel and containment shield building above approximately finished grade level for structural separation and access to penetrations for testing and inspection. The shield building and the nuclear annex are connected to form a monolithic structure.

The subsphere is that portion of the Reactor building which is below approximately finished grade level and external to the containment vessel. The subsphere houses auxiliary safety-related equipment. This area below the spherical containment allows efficient use of space for locating safety equipment adjacent to the containment vessel and eliminating excessive piping while allowing maximum access to the containment for locating penetrations.

4.10.7.3 Containment

The containment vessel is a 61 m (200 ft) diameter spherical shaped steel shell with wall thickness of approximately 19 mm (1 3/4 inch). Use of spherical steel containment provides 75% more space on the operating floor than does a typical cylindrical containment of equal volume. Allowance is made for one-piece steam generator removal.

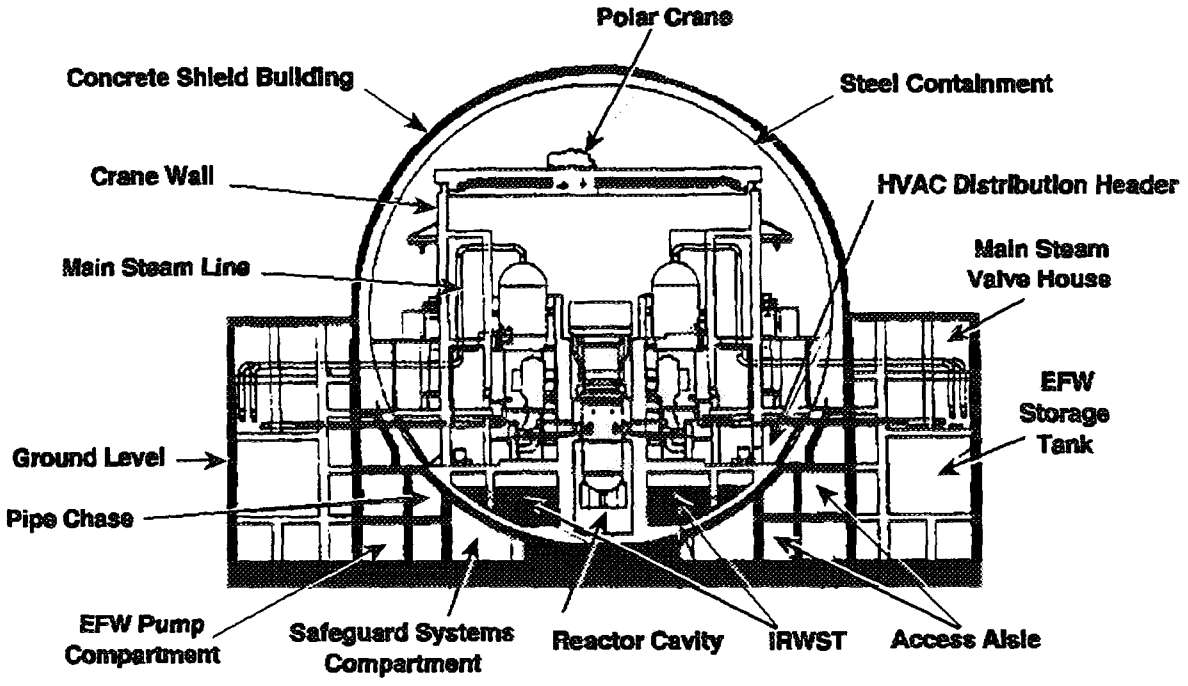


Figure 4.10-17 Containment

Containment integrity

The design basis loads are based on the peak pressure and temperature developed inside the containment as a result of a rupture in the primary coolant system up to and including a double-ended rupture of the largest pipe or a main steam line break. The containment design pressure is 365 kPa (53 psig) and the design temperature is 143°C (290°F).

The containment systems include the steel containment vessel, the containment spray system, the containment air purification and cleanup systems, the containment isolation system and the containment combustible gas control system.

The safety design basis for the containment is the requirement that the release of radioactive materials subsequent to an accident does not result in doses in excess of the values specified in 10 CFR 100. The containment must withstand the pressure and temperature of the design basis accidents without exceeding the design leakage rate of 0.5% volume for the first 24 hours, and the volume thereafter is based on a leak rate associated with half of the peak pressure assuming 0.5% volume leak rate at peak pressure.

No special provisions for protection against loss of containment integrity under external loading conditions are required. Considerations given to inadvertent operation of containment heat removal systems, such as inadvertent actuation of the containment spray system during normal unit operation, and other possible modes of plant operation that could potentially result in significant internal underpressure and external structural loading, have resulted in pressure differentials less than the design external differential pressure. The calculated minimum internal pressure amounts to -12.6 kPa (-1.83 psig), and a nominal vacuum pressure of -13.8 kPa (-2.0 psig) has been chosen for the design.

Containment integrity under severe accident conditions is addressed in Section 4.10.6.3.

4.10.7.4 Turbine building

The turbine building is a non-safety related Seismic Category II structure. The majority of the building is supported by a reinforced concrete foundation which is separated from the adjacent nuclear annex structures by a minimum gap of 15.2 cm (6 in). The turbine building superstructure lateral resisting steel frame is designed for the SSE using Seismic Category I criteria. These design requirements will prevent the collapse of the turbine building onto the adjacent nuclear annex structure.

4.10.7.5 Other buildings

Nuclear annex

The nuclear annex is composed of the control complex, diesel generator areas, main steam valve house areas, CVCS and maintenance areas, and spent fuel storage area and unit vent.

The nuclear annex is a Seismic Category I structure with exception of the unit vent and miscellaneous steel that does not support Seismic Category I equipment. These structures are Seismic Category II structures. The nuclear annex shares common walls and foundation basemat with, and is monolithically connected to, the containment shield building. In addition to the structural components, there are components designed to provide biological shielding and protection against tornado and turbine missiles. Structural components, as well as members serving as shielding components, vary in thickness from approximately 0.3 to 1.5 m (1 ft to 5 ft). In the subsphere area beneath the containment, quadrant division and physical separation of safety components virtually eliminate concerns of fire, flood, and sabotage (Figures 4.10-16 and 4.10-17).

Radwaste building

The radwaste building is a non-safety related, reinforced concrete structure classified Seismic Category II, located adjacent to the Nuclear annex. The radwaste building is separated from the adjacent nuclear annex structures by a minimum gap of 15.2 cm (6 in). Foundations and walls that house the solid and liquid waste management systems are designed such that if a safe shutdown earthquake (SSE) occurs, the maximum liquid inventory expected to be in the building will be contained. The radwaste building is designed for the SSE using Seismic Category I criteria. These design requirements will prevent the collapse of the radwaste building onto the adjacent Nuclear annex structure.

4.10.8 Technical data

General plant data

Power plant output, gross	1 389	MWe
Power plant output, net	1 350	MWe
Reactor thermal output	3 914	MWt
Power plant efficiency, net	34.3	%
Cooling water temperature	35	°C

Nuclear steam supply system

Number of coolant loops	2	
Primary circuit volume, including pressuriser	454.7	m^3
Steam flow rate at nominal conditions	2 200	kg/s
Feedwater flow rate at nominal conditions	2 200	kg/s
Steam temperature/pressure	285/6.9	°C/MPa
Feedwater temperature/pressure	232/7.2	°C/MPa

Reactor coolant system

Primary coolant flow rate	20 800	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	291	°C
Coolant outlet temperature, at RPV outlet	323.9	°C
Mean temperature rise across core	32.9	°C

Reactor core

Active core height	3.810	m
Equivalent core diameter	3.658	m
Heat transfer surface in the core	6 590	m^2
Fuel weight	116.6	tU
Average linear heat rate	18.1	kW/m
Average fuel power density	33.7	kW/kg U
Average core power density (volumetric)	95.5	kW/l
Thermal heat flux, F_q	602	kW/m^2
Enthalpy rise, F_{Π}	46.1	cal/kg

Fuel material	Sintered UO_2, or PuO_2	
Fuel assembly total length	4 350	mm
Rod array	square, 16×16	
Number of fuel assemblies	241	
Number of fuel rods/assembly	236	
Number of control rod guide tubes	5	
Number of spacers	11	
Enrichment (range) of first core	1.9/2.8/3.3	Wt%
Enrichment of reload fuel at equilibrium core		Wt%
Operating cycle length (fuel cycle length)	18-24	months
Average discharge burnup of fuel, first core	31 700	MWd/t
Cladding tube material	Zr 4	
Cladding tube wall thickness	0.635	mm
Outer diameter of fuel rods	9.7	mm
Overall weight of assembly	662.7	kg
Active length of fuel rods	3 810	mm
Burnable absorber, strategy/material	Er_2O_3	
Number of control rods (control element assemblies)	93	
Absorber rods per control assembly	4 or 12	
Absorber material	B_4C in 48 CEAs; Ag-In-Cd in 20 CEAs; and Inconel 625 in 25 CEAs	
Drive mechanism	Magnetic jack	
Positioning rate	762	mm/min
Soluble neutron absorber	Boron	

Reactor pressure vessel

Cylindrical shell inner diameter	4 630	mm
Wall thickness of cylindrical shell	229	mm
Total height	15 280	mm
Base material: cylindrical shell	SA-509	
RPV head	low alloy carbon steel	
liner	stainless steel	
Design pressure/temperature	17.2/343.3	MPa/°C

Transport weight (lower part) [include RPV head]	508	t
RPV head	81.2	t

Steam generators

Type	vertical U-tube with integral moisture separator and economizer	
Number	2	
Heat transfer surface	14 660	m^2
Number of heat exchanger tubes	12 580	
Tube dimensions	1.07	mm
Maximum outer diameter	5 890	mm
Shell side, design pressure/temperature	8.3/298.9	MPa/°C
operating pressure	7.6	MPa
Maximum moisture content at SG outlet	0.25	%
Total height	23 000	mm
Transport weight	748	t
Shell material, primary side	low alloy steel clad with stainless steel	
secondary side	carbon steel	
Tube material	SB-163 NiCr Fe alloy	

Reactor coolant pump

Type	Vertical, single-stage, centrifugal	
Number	4	
Design pressure/temperature	17.2/343.3	MPa/°C
Design flow rate (at operating conditions) [7 m^3/s =]	5 200	kg/s
Pump head	114	mWG
Power demand at coupling, cold/[hot]	8 950	kW
Pump casing material (base metal/clad)	low alloy carbon steel/ stainless steel	
Pump speed	1 200	rpm

Pressuriser

Total volume	68	m^3
Steam volume: full power/[zero power]	34	m^3
Design pressure/temperature	17.2/371.1	MPa/°C
Heating power of the heater rods	2 400	kW
Number of heater rods	48	
Inner diameter	4 630	mm
Total height	14 800	mm
Material (base metal/clad)	low alloy carbon steel/stainless steel	
Transport weight	511	t

Pressuriser relief tank

Not applicable

Total volume	m^3
Design pressure/temperature	MPa/°C
Inner diameter (vessel)	mm
Total height	mm
Material	
Transport weight	t

Primary containment

Type	Dry, dual containment	
Overall form (spherical/cyl.)	spherical steel surrounded by reinforced concrete shield building	
Dimensions (diameter/height), containment vessel	61	m
shield building	65.8	m
Free volume	96 300	m^3
Design pressure/temperature (DBEs)	365/143.3	kPa/°C
(severe accident situations)	≤ ASME Service Level C for 24 hours/-	kPa/°C
Design leakage rate	< 0.5	vol%/day
Is secondary containment provided?	yes, by the shield building and containment shield annulus	

Reactor auxiliary systems

Reactor water cleanup;	CVCS letdown/charging capacity [26 m^3/h]	7.2	kg/s
	filter type		
Residual heat removal,	at high pressure design flow/division [1 136 m^3/h]	315.5	kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure/pump [185 m^3/h]	51.4	kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage	230/22.8	kV
	rated capacity	3×760	MVA

Plant transformers,	rated voltage		kV
	rated capacity		MVA
Start-up transformer	rated voltage		kV
	rated capacity		MVA
Medium voltage busbars (4.16 kV)		4 safety/4 non-safety	
Number of low voltage busbar systems (480 V)		4 safety/12 non-safety	
Standby diesel generating units: number		2	
	rated power	6.67	MW
Number of diesel-backed busbar systems (divisions/buses)		2/4	
Voltage level of these		4 160	V ac
Number of DC distribution centers		5	
Voltage level of these		250 (2), 125 (3)	V dc
Number of battery-backed AC safety busbar systems		6	
Voltage level of these		120	V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	Tandem-compound	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/ 3 LP	
Turbine speed	1 800	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	7.2/287.8	MPa/°C

Generator

Type	H_2 inner cooled	
Rated power	1 573	MVA
Active power	1 415.7	MW
Voltage	24	kV
Frequency	60	Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type	Three shell, three pass, divided water boxes	
Number of tubes		
Heat transfer area	99 460	m^2
Heat transfer capacity, design	2 473	MW
Cooling water flow rate	37.9	m^3/s
Cooling water temperature	35 (max)	°C
Condenser pressure, design: shell	9.9/9.7/8.1	kPa
water box	1.73	MPa

Condensate pumps

Number	3	
Flow rate [2 839 m^3/h]	788.6	kg/s
Pump head		
Temperature		°C
Pump speed		rpm

Condensate cleanup system

Full flow/part flow	full flow
Filter type	cation and mixed bed ion exchanger

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/ °C

Feedwater pumps

Number	3	
Flow rate		kg/s
Pump head		
Feedwater temperature	232.2	°C
Pump speed	varable	rpm

Condensate and feedwater heaters

Number of heating stages	6
Redundancies, low pressure	3 parallel paths
high pressure,	2 parallel paths

4.10.9 Project status and planned schedule

The NRC has completed its review of the System 80+ Standard Plant Design, approving advanced design features and closing all design-related safety issues. Final Design Approval was granted in July, 1994.

The System 80+ design is proceeding through the Design Certification Rulemaking process. The NRC issued the System 80+ Notice of Proposed Rulemaking (NOPR) on April 7, 1995. As a result of comments received, the NRC revised the proposed rule and issued a supplement to the NOPR in April, 1996. Comments focused on policy and process issues regarding the proposed rule rather than on features of the System 80+ design that the rule would codify. Design Certification of System 80+ is expected in early 1997[2].

References

Combustion Engineering Standard Safety Analysis Report - Design Certification (CESSAR-DC), ABB Combustion Engineering, 2000 Day Hill Rd., Windsor, CT 06095, USA

NUREG-1462, Vol.1, Final Safety Evaluation Report Related to the Design Certification of the System 80+ Design, Docket No. 52-002, Published August, 1994, US. Nuclear Regulatory Commission, Washington, DC 20555, USA

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Technical Paper, System 80+TM PRA Insights on Severe Accident Prevention and Mitigation, D. J. Finnicum, M. C. Jacob, R. E. Schneider, and R. A. Weston, ABB Combustion Engineering, Presented at 4th International Topical Meeting on Nuclear Thermal Hydraulics, Operations and Safety, April 6-8, 1994, Taipei, Taiwan

Technical Information on Design Features of the ABB Combustion Engineering's System 80+TM Standard Plant Design, Revised - July 15, 1994, ABB Combustion Engineering, 2000 Day Hill Rd., Windsor, CT 06095, USA

[2] Subsequently Design Certification by the USNRC of the System 80+ has been achieved in May 1997.

4.11 WWER-1000 (V-392), ATOMENERGOPROJECT/GIDROPRESS, RUSSIAN FEDERATION

4.11.1 Introduction

The principle of ensuring the safety of the personnel, the population and the environment against radiation effects exceeding the prescribed radiation doses is used as the basis for design. This principle also addresses the standards for releases of radioactive substances and their content in the environment under normal operation conditions, anticipated operational occurrences, and in design and beyond-design-basis accidents during the plant service life.

The objective of the design of the reactor plant and of the nuclear plant process systems was to achieve that the estimated probability of a severe fuel damage does not exceed 1.0E-6 per reactor-year and that the probability of accidental radioactive releases prescribed by regulatory requirements does not exceed 1.0E-7 per reactor-year.

Nuclear power plant (NPP) safety is achieved by consistent implementation of the principle of "defence-in-depth" based on the application of a system of barriers on the path of spreading ionizing radiation and radioactive substances into the environment, as well as on a system of engineered safeguards and organizational provisions for protection of these barriers.

A consistent implementation of the "defence-in-depth" principle is provided with the following:

- installation of successive physical barriers on the path of spreading radioactive substances: fuel matrix, fuel element cladding, primary circuit boundary, containment;
- taking into account postulated initial events that could lead to a loss of efficiency of those barriers;
- determination, for each postulated event, of design measures and actions of operating personnel required to keep the integrity of the barriers mentioned, and mitigation of consequences of damage to such barriers;
- minimization of the probability of accidents resulting in a release of radioactive substances;
- consideration of beyond design basis accident management.

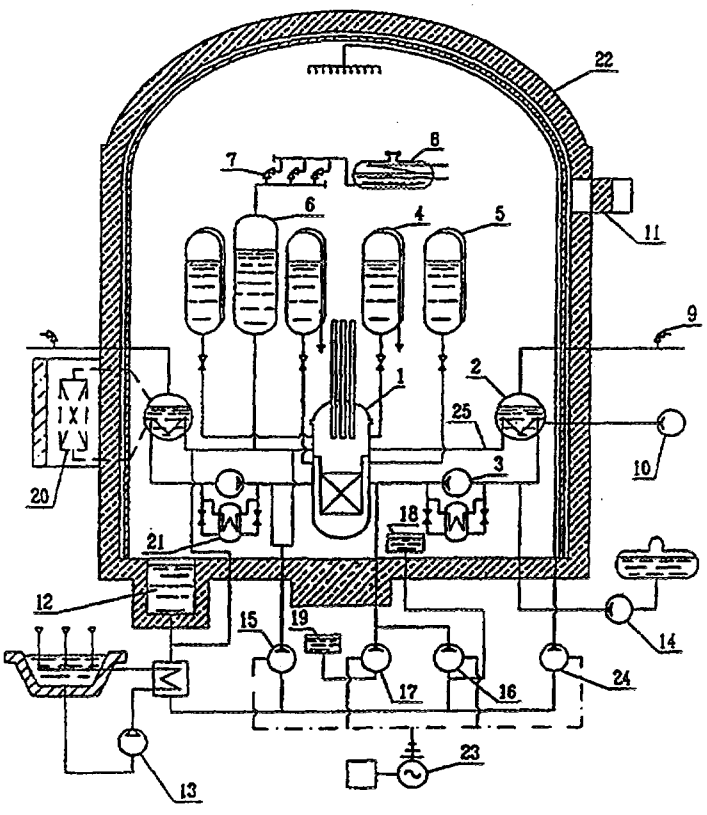
The principal technical decisions are supported by the operational experience from more than 90 reactor-years of WWER-1000-type NPPs.

The design is developed in accordance with the latest versions of the Russian "Safety Regulations for NPPs" (References [1] and [2]) by three organizations: OKB "Gidropress", Russian National Research Centre "Kurchatov Institute" and LOAEP, all being well known designers of WWER NPPs. QA requirements of the IAEA and international standards ISO 9000 are taken into account in the design.

In the plant safety concept, modern worldwide trends in NPP safety improvements are considered in order to meet the normative requirements for NPP safety, which are constantly becoming more strict, for as long a period as possible.

The principal features that largely determine nuclear plant safety are as follows:

- possibility of providing subcriticality with solid control rods at any moment of the plant life under a coolant temperature decrease to 120°C;
- application of horizontal steam generators with a large water inventory and with better conditions for natural circulation in the primary circuit in comparison with vertical steam generators;
- application of an emergency core cooling system, based on the principles of both passive and active operation, that provides for the possibility of long-term residual heat removal after accidents with primary leaks accompanied by a station blackout;



1. Reactor
2. Steam generator
3. Main pump
6. Pressurizer
7. Pressurizer safety valve
8. Bubbler
9. SG safety valve
10. SG emergency feedwater pump
11. Filter
12. Boron solution store
13. Service water pump
14. Primary makeup pump
15. ECCS LP pump
16. ECCS HP pump
17. ECCS HP pump
18. Boron solution
19. Boron solution
20. Passive residual heat removal system
21. Rapid boron supply system
22. Double containment
23. Diesel generator
24. Sprinkler pump
25. Main pipe

Figure 4.11-1 V-392 reactor plant - major systems

- application of a passive core flooding system;
- application of a quick boron supply system;
- application of a double wall concrete containment;
- application of a diagnosis system for the equipment of the systems important to safety for periodical inspection during shutdown, and for on-line diagnosis during reactor operation;
- application of an automatic control system of improved reliability with self-diagnosis, and of an expert system giving advice to the operator;
- application of an emergency system for discharging and purification of radioactive materials of the steam-gas mixture vented from the containment if the pressure exceeds the allowable values in beyond design basis accidents.

4.11.2 Description of the nuclear systems

4.11.2.1 Primary circuit and its main characteristics

A schematic drawing of the reactor plant and the major reactor plant systems is shown in Figure 4.11-1.

The reactor plant includes a reactor coolant system, a primary pressure control system and a primary overpressure protection system. The reactor coolant system consists of 4 loops, with a horizontal steam generator and a reactor coolant pump in each one. The primary pressure control system includes a pressurizer, pressurizer spray valves, a bubbler, safety valves, pressurizer steam-gas mixture removal valves and throttling device, and connecting pipelines.

The bubbler is a horizontal cylindrical vessel with elliptic bottoms filled with water to 2/3 of its volume. Two safety membranes are installed in its cylindrical part. Steam discharge headers with nozzles and a heat exchanger are arranged inside the bubbler.

There are three pressurizer spray lines in parallel; two of them which have an inner diameter of 125 mm, are equipped with shut-off valves, whereas the third, with a diameter of 105 mm, is provided with control valves. The pressurizer spray lines are connected to the primary loop makeup water pump line. The discharge from the safety valves goes via a pipeline into the bubbler.

The steam-gas mixture removal line with valves and throttling device connects the pressurizer steam volume with the bubbler; this line operates during plant startup when the primary coolant is heated, and in some accident situations.

In the event of a primary system pressure increase beyond the operation set point of the pressurizer spray, primary coolant from the cold leg of the primary loop is injected through the pressurizer spray line and the nozzles in the pressurizer steam volume. The spray water can also be provided by the primary loop makeup water pumps in case the main coolant pumps have stopped operating. The steam and water volumes of the pressurizer are determined in such a way that the set points of the safety valves will not be exceeded in any design basis condition (including design basis accident situation), even if the pressurizer spray fails to function.

4.11.2.2 Reactor core and fuel design

The WWER-1000 concept is based on an evolutionary approach, and the core design reflects the operation experience of current reactors. The core is designed to have stabilizing reactivity coefficients under all operating conditions to meet the basic safety objectives. Reactivity control is accomplished by changing of the boron concentration in the primary coolant and by moving control rod assemblies. As a rule, slow reactivity changes due to changes in burnup and amount of xenon are compensated by changes of the boron concentration, while rapid reactivity changes for adaptation of the power level to the load are accommodated by control rod insertions or withdrawals.

The power level of the reactor is monitored by means of ex-core instrumentation, but the 3-D power distribution in the core is determined by an in-core instrumentation system based on self-powered measuring detectors.

The reactor core consists of 163 fuel assemblies. Each fuel assembly comprises 311 fuel rods and 18 guide tubes; their active length is 3.53 m. The fuel cladding is made of zirconium alloy tubing which contains sintered UO_2 pellets with an initial enrichment that does not exceed 4-4.4 weight %. The average linear heat rate of the fuel rods amounts to 16.67 kW/m. There are 121 control rod clusters which are also used by the reactor scram system for rapid shutdown of the reactor. Pitch electromagnetic drives with position indicators are used as driving devices for the control rod clusters. The drives are installed on the reactor vessel head.

The effective operation time between refuellings is 7000 effective full power hours. The average burnup of the unloaded fuel is up to 43 MW days/kg U. The number of fresh assemblies loaded during annual refuelling is 54.

4.11.2.3 Fuel handling and transfer systems

The fuel handling and transfer systems are intended for loading of fresh fuel assemblies and replacing spent fuel assemblies, shutdown absorber rods and burnable poison absorber rods. Fuel handling and transfer are performed under water for radiation protection. For fuel handling to take place, the reactor is shut down and depressurized, the head of the reactor pressure vessel removed and the control rod guide tube unit extracted from the reactor vessel, and the concrete well above the reactor and the fuel pool filled with water.

The fuel handling and transfer systems include the following compartments with necessary equipment: the reactor concrete well; the fuel pool; and the transfer compartment

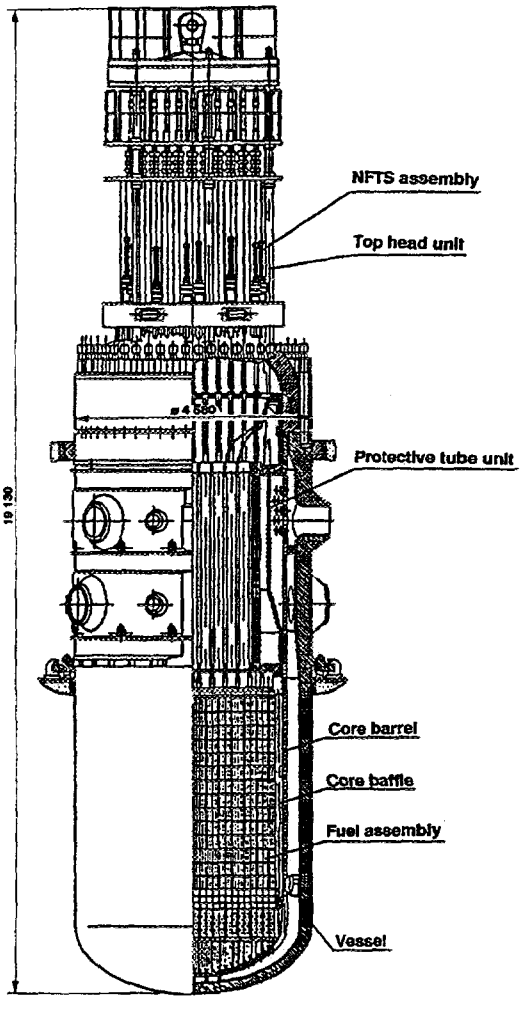


Figure 4.11-2 V-392 reactor assembly

The fuel pool is located in the vicinity of the reactor concrete well; they are connected by a transfer channel designed for transportation of one fuel assembly at a time. The fuel pool is provided with storage structures for spent fuel assemblies; these structures consist of separate sections designed for storing of fuel assemblies and sealed containers for failed fuel assemblies.

The transfer compartment is provided with an universal nest for location of a plant-internal transportation unit for fresh fuel assemblies and for a transport cask for spent and decayed fuel assemblies. The transfer compartment is connected to the fuel pool by the transfer channel through which loading of fresh fuel assemblies take place. The fresh fuel assemblies are taken out of the plant-internal transportation unit by the refuelling machine and installed in the core in accordanse with the core refuelling chart.

After unloading from the core, the spent fuel assembly is placed in a container with a fuel assembly defect monitoring system, and then transported to the storage structure or the sealed container for spent and decayed fuel assemblies in the fuel pool depending on the results of the defect inspection.

Manipulations for control rod assembly replacements are similar to those for fuel handling; control rod clusters and burnable poison rods extracted from the reactor core are installed in empty fuel assemblies or in storage structures in the fuel pool for storage.

The refuelling machine handles only one fuel assembly, one control rod cluster, or one fuel assembly with the control rod cluster inside it, at a time. Heat removal from the reloaded fuel assemblies is accomplished by the fuel pool cooling system.

4.11.2.4 Primary components

Reactor pressure vessel

A schematic drawing of the reactor assembly is shown in Figure 4.11-2. The reactor vessel is similar to that of a serial WWER-1000 reactor.

The reactor vessel consists of a flange, upper and lower shells of nozzle zone, support shhell, cylindrical shell and elliptic bottom, welded together by circumferential seams. The vessel has two rows of nozzles of 850 mm inner diameter (four nozzles in each row); there are two nozzles for the pipeline connections of the ECCS hydroaccumulators on each row level, and there is one instrumentation penetration nozzle in the upper row. The inner surfaces of the vessel and the nozzles are plated with a corrosion-resistant layer. A separation ring is welded to the vessel inside between the upper and lower nozzle rows for separation between the inlet and outlet coolant flows, and a support rib is made on the vessel support shell for vessel attachment on the support structure.

The vessel cover consists of a truncated ellipsoid and a flange that are connected with eachother by a circumferential weld seam. Penetrations for in-vessel instrumentation, nozzles for gas removal lines, and holes for control rod housings are provided on the cover ellipsoid.

Reactor internals

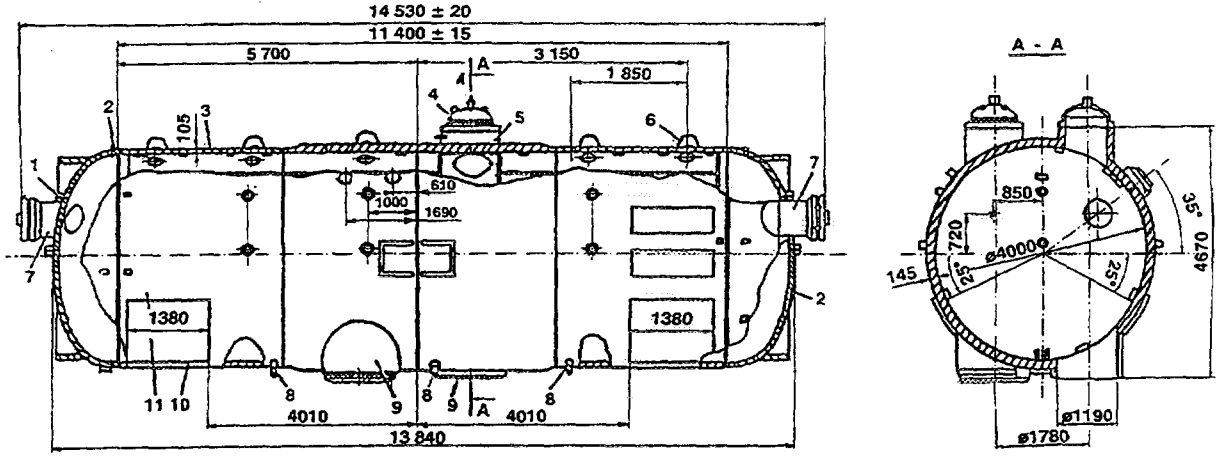
No information provided.

Steam generators

The steam generator (SG) is of the horizontal, single-vessel type, with an immersed heat exchange area consisting of tube bundles arranged horizontally. The SG is a modernization of the standard SG PGV-1000. The positive experience from operating WWER-1000 and WWER-440 SGs has been taken into account. In particular, the perforated part of the primary collector is made of stainless steel 0KH18N10T that has shown good properties during the operation of the WWER-440 SG primary collectors. For internals inspection, hatches of 500 mm diameter on the elliptic bottom, as well as hatches of 1000 mm diameter in the cylindrical part of the steam generator, are provided. A schematic drawing of the steam generator is shown in Figure 4.11-3.

Pressurizer

For the V-392 plant it is anticipated to use the pressurizer (Figure 4.11-4) applied in the standard WWER-1000 design



1. Sleeve with Dnom 20
2. End plate
3. Shell
4. Top plate
5. Nozzle with Dnom 800
6. Nozzle with Dnom 350
7. Nozzle with Dnom 500
8. Periodic blowdown nozzle
9. Nozzle with Dnom 1200
10. Support plate
11. Strap

Figure 4.11-3 V-392 steam generator

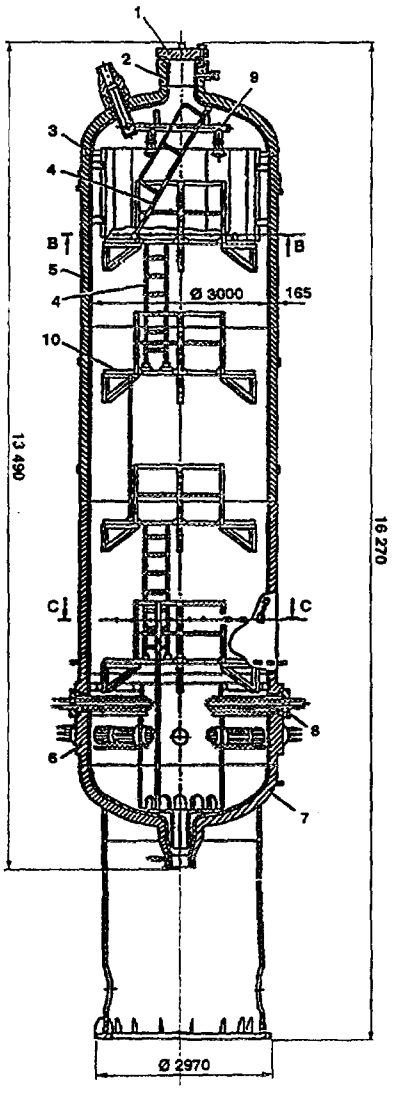


Figure 4.11-4 V-392 pressurizer

The pressurizer with primary pressurization system is designed for maintaining primary system pressure within acceptable range in all reactor plant design conditions. The pressurizer proper is a vertical vessel mounted on a cylindrical support structure. There are nozzles for spray pipelines and safety valve steam discharge line on the upper end, and a nozzle for connection between the pressurizer and the hot leg is provided in the vessel bottom.

Nozzles for level gauges and casings for thermo-resistors measuring coolant temperature inside the vessel are located on its cylindrical shell.

A spray device and a tubular electric heater are located inside the pressurizeer. The spray device is intended for water spraying in the steam volume and condensation of steam; it is made as a discharging header fastened to the top of the vessel. The inside surface of the pressurizer is plated with a corrosion resistant material. All pressurizer internals are made of austenitic stainless steel.

Reactor coolant pumps

The reactor coolant pump (RCP) (Figure 4.11-5) is a vertical, single-stage, centrifugal pump with an autonomous lubrication system housed in a spherical case. The RCP subsystems prevent the escape of radioactive coolant out of the primary system. The electrical motor is of the vertical type, three-phase, and with two speeds. A non-combustible lubricant is used in the electrical motor.

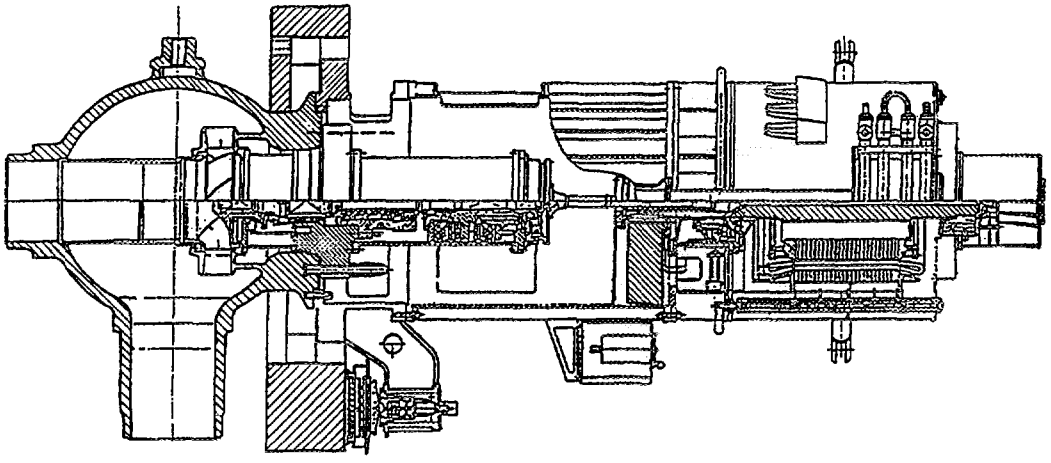


Figure 4.11-5 V-392 reactor coolant pump

Main coolant lines

The reactor pressure vessel nozzles for the coolant inlet and outlet (the cold and hot legs) have an inner diameter of 850 mm.

4.11.2.5 Reactor auxiliary systems

Chemical and volume control system

No information provided.

4.11.2.6 Operating characteristics

No information provided.

4.11.3 Description of turbine generator plant system

4.11.3.1 Turbine generator plant

No information provided.

4.11.3.2 Condensate and feedwater systems

No information provided.

4.11.3.3 Auxiliary systems

No information provided.

4.11.4 Instrumentation and control systems

4.11.4.1 Design concepts including control room

The reactor plant control system secures fulfillment of the following main functions:

- monitoring of the unit operation, radiological situation, state of equipment and systems in all conditions;
- remote control;
- automatic control of reactor plant, secondary circuit and auxiliary systems parameters;
- process protection and interlocking;
- emergency and preventive protection of the reactor.

The following priority of control commands (in the order of decreasing priority) is secured in the control system of the reactor plant:

- emergency and preventive reactor protection and control of safety systems;
- process protection and interlocking;
- manual remote control;
- automatic control (the main controlled parameters are: neutron flux in the core, primary pressure, secondary pressure, water levels in steam generators and the pressurizer);
- recording and archiving of the main parameters under normal and emergency conditions.

Under emergency conditions the main parameters and the equipment status are continuously recorded.

In case of a main control room (MCR) failure, for example during a fire, the reserve control room (RCR) is used to provide:

- reactor shutdown;
- monitoring of subcriticality;
- reactor cool-down;
- putting into operation of confining systems.

Possibility of control of the systems important for safety is retained from RCR. Ensurance of habitability under loss of regular ventilation systems during a safe shutdown earthquake (SSE) and associated fire, or other destructions on the site, is provided for the reserve control room.

Local control panels which do not require interaction with the MCR and the RCR are provided. Their existence, in a number of cases, is determined by considerations of NPP layout. The RCR is provided with an access admittance check system.

4.11.4.2 Reactor protection system and other safety systems

The following systems belong to the class of automatic safety systems:

- reactor emergency protection system
- primary overpressure protection system
- emergency core cooling system
- system of passive heat removal from the reactor plant
- passive hydrostatic core flooding system
- system of quick-acting isolation valves in steamlines
- secondary overpressure protection system
- quick-acting boron supply system
- emergency diesel-generators
- emergency system of reliable direct and alternate electric current power supply

Reactor emergency protection system

The reactor emergency protection system provides reliable disconnection of electric power supply and, as a consequence, a drop of emergency protection rods into the core. In this case, disappearance of signal of original cause does not stop the initial action of the emergency protection.

The solid rods of the emergency protection actuate in response to the following signals:

- decrease of reactor period
- increase of neutron flux
- decrease of margin to saturation temperature in any hot leg
- increase of coolant temperature in any hot leg
- decrease of pressure differential over the primary coolant pumps
- de-energization of several primary coolant pumps
- decrease of pressure in the reactor
- increase of pressure in the reactor
- increase of pressure in a SG
- decrease of pressure in a SG coinciding with a definite increase of the primary and secondary saturation temperature difference
- decrease of water level in a SG
- increase of pressure in the containment

The parameters chosen permit to secure the necessary reduction of reactor power for meeting the design criteria under all design conditions. Automatic disconnection of power governors, and interlocking of all operator's actions on control rods occur when the emergency protection operates.

Two sets of instrumentation are provided, generating the commands for the emergency protection and operating in parallel using an "or" logic. The signals for operation of the emergency protection are generated using a "2-out-of-3" majority logic in any set.

However, with the aim of enhancing of NPP safety, failure of the emergency protection system of the reactor is postulated in some beyond design basis accidents by considering scram failure under operational occurrences and in design accidents.

4.11.5 Electrical systems

4.11.5.1 Operational power supply systems

No information provided.

4.11.5.2 Safety-related systems

Diesel generators

The diesel generators provide the power supply to safety related systems for 2 days using the internal fuel stock, and for unlimited time if fuel is provided from the outside.

System of reliable direct current power supply

The system of reliable direct current power supply consists of storage batteries. It provides power to the electromagnetic circuits for actuating the automatic safety systems as well as for recording necessary plant parameters during 24 hours.

4.11.6 Safety concept

4.11.6.1 Safety requirements and design philosophy

No information provided.

Deterministic design basis

No information provided.

Risk reduction

No information provided.

External and internal hazards

No information provided.

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

Safety systems configuration

No information provided

Safety injection systems

No information provided

In-containment refuelling water storage tank

No information provided

Emergency feedwater system

No information provided

Residual heat removal system

No information provided.

Emergency core cooling system

The emergency core cooling system (ECCS) has been designed to provide a possibility of long-term residual heat removal in case of primary leak accidents accompanied by a station blackout. The ECCS involves two subsystems:

- a passive subsystem using hydro-tanks with nitrogen under pressure; and
- an active subsystem with high pressure injection pumps.

The hydro-tanks with nitrogen under pressure will provide the coolant injection during the first stage of such an accident, and the active subsystem takes over when the hydro-tanks have been emptied.

The active part of the ECCS includes two independent trains having an overall redundancy within each train. Each of the 4 subtrains thus formed is capable to fulfill the necessary system functions. Each subtrain includes the sump of the containment, a high pressure injection (HPI) pump, a jet pump installed on the discharge side of the HPI pump, an emergency cooling-down heat exchanger and pipelines and fittings.

The energy supply for the active elements of the system is provided by the reliable emergency electric power supply system. Each of the four subtrains of the system has its own subtrain of reliable electric power supply, including a diesel-generator.

Passive residual heat removal from the reactor plant

A passive residual heat removal system (PHRS) is included in the design to remove heat from the reactor plant. The design basis of the PHRS is that in case of a station blackout, including loss of emergency power supply, the removal of residual heat should be provided without damage of the reactor core and the primary system boundary for at least 24 hours. The PHRS consists of four independent trains, each of them is connected to the respective loop of the reactor plant via the secondary side of the steam generator. Each train has pipelines for steam supply and removal of condensate, valves, and an air-cooled heat exchanger outside the containment. Part of the PHRS is an air-cooled heat exchanger that is installed outside of the containment. The steam that is generated in the steam generators due to the heat released in the reactor, condenses and rejects its heat to the ambient air. The condensate is returned back to the steam generator. The motion of the cooling medium takes place in natural circulation.

Passive core flooding system

The passive core flooding system includes four groups of hydro-tanks under atmospheric pressure which are coupled with the pipelines connecting the ECCS hydro-accumulators with the reactor. The hydro-tanks of the passive core flooding system are connected to the primary system at 1.5 MPa and allow to flood the core due to the hydrostatic pressure of the water column, and to remove the reactor residual heat in the last stage of a LOCA for at least 12 hours.

Quick boron supply system

The quick boron supply system (QBSS), being developed as an additional reactor trip system, comprises a system of 4 special loops bypassing the main coolant pumps. Each loop consists of a hydro-accumulator containing concentrated boron acid solution, and of pipelines with quick-acting valves that do not require electric power for their opening. These valves open during occurrences and accidents with failure of scram, and concentrated boron solution is pressed out of the hydro-accumulators into the primary loops, and further into the reactor. In case of a station blackout the boron solution delivery occurs in the period of reactor coolant pump (RCP) coast-down. The RCPs have a considerable flywheel inertia which provides the possibility of ejecting all boron concentrate from the QBSS hydro-accumulators. The amount and concentration of the boron solution are chosen to provide a definite equivalency from the viewpoint of reactivity inserted by this system and by the solid absorber scram. In fact, this system, being part of the primary coolant circulation system, allows to consider a reactor plant with such a system as a plant with increased "inherent" safety.

Primary overpressure protection system

The system comprises three safety valves for discharging steam or a steam-water mixture from the pressurizer if its pressure increases above the permissible level, as well as a subsystem for receiving a steam-water mixture. This subsystem involves a bubbler and pipelines connecting it to the outlets of the safety valves.

Secondary overpressure protection system

The secondary overpressure protection system is intended for preventing the secondary pressure to increase above the permissible value. The system incorporates quick-acting steam dumping valves and steam generator safety valves.

System of quick-acting isolation valves in steam lines

Quick-acting isolation valves in the steam lines close at:

- increase of water level in the SGs above the permissible level; and
- increase of radioactivity in the SGs above the permissible level, on the appearance of signals of a steamline rupture.

They are intended, respectively, for the protection of the turbine from steam of high humidity, for preventing radioactivity releases from the SGs, and for restricting the steam blow down after a rupture of the secondary circuit.

4.11.6.3 Severe accidents (Beyond design basis accidents)

Severe accident mitigation strategy

No information provided.

Severe accident prevention and mitigation features

No information provided.

4.11.7 Plant layout

4.11.7.1 Buildings and structures, including plot plan

Design requirements

Physical separation aspects

Radiation protection aspects (accessibility, shielding, ventilation)

Loads acting upon the outer protective shell of the containment

The design is performed taking into account two levels of seismicity: the operating basis earthquake (OBE) of magnitude 7 on the MSK-64 scale and the safe-shutdown-earthquake (SSE) of magnitude 8 on the MSK-64 scale.

The reactor plant equipment is calculated for seismic effects proceeding from the following conditions. During an operating basis earthquake normal operation of the reactor plant is to be provided. During a safe-shutdown-earthquake reactor shutdown and reactor plant shutdown cooling are to be provided.

All civil structures, process and electrotechnical equipment, pipelines, instrumentation and so on are divided into 3 seismic categories depending upon the degree of responsibility for safety ensurance during seismic effects and for serviceability after an earthquake. Components and systems being related to seismic category 1 (the highest) shall fulfill their functions concerning NPP safety ensurance in the course of an earthquake, and after it, at intensities up to SSE, and at OBE to they shall keep their serviceability. Seismic category systems include:

- systems of normal operation, failure of which during seismic events, SSE inclusive, may result in a release of radioactive material in such quantities that causes excessive population dose in comparison with the specified values; and
- safety systems for keeping the reactor in a subcritical state, for emergency heat removal from the reactor, for confinement of radioactive products, and buildings, structures and equipment, mechanical damages of which during seismic events, SSE inclusive, may result in failure of these systems.

The outer protective shell structure is designed for the impact of an environmental shock wave having a front pressure of 0.03 MPa, and a compression phase duration of up to 1 second, and for a crash of a 5.0 tonne plane creating a 1200 ts impulse with an impact duration time equal to 0.1 second and with a contact area equal to 12.0 square meters.

Loads on the inner containment

The inner containment is designed for the following loads:

- impact of the maximum design basis accident conditions with a maximum excess pressure of 0.4 MPa, and a maximum temperature of 150°C;
- impact of missiles and steam-water jets inside the containment.

Under design basis accidents, the isolation systems provide for confinement of radioactive releases inside the containment and for heat removal from the containment. For beyond design basis accidents, a system for containment pressure venting and a filtered discharge from the containment is provided.

4.11.7.2 Reactor building

No information provided.

4.11.7.3 Containment

A double wall containment is provided in the design. The inner shell bears the loads arising from a sequence of internal accidents. The outer shell provides protection from external loads (as tornado, hurricane, shock wave, plane crash etc.).

Containment integrity

No information provided.

Corium retention

No information provided.

4.11.7.4 Turbine building

No information provided.

4.11.7.5 Other buildings

Safeguard and fuel buildings

4.11.8 Technical data

General plant data		
Power plant output, gross		MWe
Power plant output, net		MWe
Reactor thermal output	3 000	MWt
Power plant efficiency, net		%
Cooling water temperature		°C
Nuclear steam supply system		
Number of coolant loops	4	
Primary circuit volume, including pressuriser		m^3
Steam flow rate at nominal conditions [4x1470 t/h]	1 633	kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	/6.27	°C/MPa
Feedwater temperature/pressure	220/	°C/MPa
Reactor coolant system		
Primary coolant flow rate [24.444m^3/s]	18 040	kg/s
Reactor operating pressure	15.7	MPa
Coolant inlet temperature, at RPV inlet	293.9	°C
Coolant outlet temperature, at RPV outlet	323.3	°C
Mean temperature rise across core		°C
Reactor core		
Active core height	3.53	m
Equivalent core diameter	3.16	m
Heat transfer surface in the core	4 957	m^2
Fuel weight [UO_2]	74	t
Average linear heat rate	16.67	kW/m
Average fuel power density	45.81	kW/kg U
Average core power density (volumetric)	107.5	kW/l
Thermal heat flux, F_q	605.2	kW/m^2
Enthalpy rise, F_H		

Fuel material	Sintered UO_2	
Fuel assembly total length	4 670	mm
Rod array	triangle	
Number of fuel assemblies	163	
Number of fuel rods/assembly	311	
Number of guide tubes for control rods/instr.	18/1	
Number of spacers		
Enrichment (range) of first core	1.6/ 3/ 4.4	Wt%
Enrichment of reload fuel at equilibrium core	4.4	Wt%
Operating cycle length (fuel cycle length)	12	months
Average discharge burnup of fuel	43 000	MWd/t
Cladding tube material	zirconium alloy	
Cladding tube wall thickness	0.61	mm
Outer diameter of fuel rods	9.1	mm
Overall weight of assembly		kg
Active length of fuel rods	3 530	mm
Burnable absorber, strategy/material		
Number of control rod assemblies	121	
Absorber rods per control assembly	18	
Absorber material	B_4C	
Drive mechanism	Magnetic jack	
Positioning rate		mm/min (or mm/s)
Soluble neutron absorber	boron	
Reactor pressure vessel		
Cylindrical shell inner diameter	4 070	mm
Wall thickness of cylindrical shell	190	mm
Total height	19 100	mm
Base material: cylindrical shell	15Kh2NMFA	
RPV head		
liner	stainless steel	
Design pressure/temperature	17.65/350	MPa/°C
Transport weight (lower part) [include. head]	417	t
RPV head		t

Steam generators

Type	horizontal	
Number	4	
Heat transfer surface	5 130	m^2
Number of heat exchanger tubes	9 157	
Tube dimensions	16x1.5	mm
Maximum outer diameter	4 300	mm
Total height	9 500	mm
Transport weight	300	t
Shell and tube sheet material	10GN2MFA/0Kh18N10T	
Tube material	08Kh18N10T	

Reactor coolant pump

Type	Single-stage, centrifugal pump	
Number	4	
Design pressure/temperature	17.6/350	MPa/°C
Design flow rate (at operating conditions) [6.111m^3/s]	4 510	kg/s
Pump head		
Power demand at coupling, cold/hot	6 700/5 000	kW
Pump casing material	Stainless steel	
Pump speed	1000/750	rpm

Pressuriser

Total volume	79	m^3
Steam volume: full power/zero power	24/-	m^3
Design pressure/temperature	17.65/350	MPa/°C
Heating power of the heater rods	2 520	kW
Number of heater rods		
Inner diameter	3 000	mm
Total height	13 000	mm
Material	10GN2MFA	
Transport weight	214	t

Pressuriser relief tank

Not applicable

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Dry, double wall	
Overall form (spherical/cyl.)	cylindrical, in steel/reinforced concrete	
Dimensions (diameter/height)	53/61.6	m
Free volume	75 000	m^3
Design pressure/temperature (DBEs)	500/	kPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate, inner shell	0.3	vol%/24 h
outer shell	15	vol%/24 h
Is secondary containment provided?	yes, space between the walls	

Reactor auxiliary systems

Reactor water cleanup,	capacity	kg/s
	filter type	
Residual heat removal,	at high pressure	kg/s
	at low pressure	kg/s
Coolant injection,	at high pressure	kg/s
	at low pressure	kg/s

Power supply systems

Main transformer,	rated voltage	kV
	rated capacity	MVA
Plant transformers,	rated voltage	kV
	rated capacity	MVA
Start-up transformer	rated voltage	kV
	rated capacity	MVA
Medium voltage busbars (6 kV or 10 kV)		
Number of low voltage busbar systems		
Standby diesel generating units: number		
	rated power	MW
Number of diesel-backed busbar systems		
Voltage level of these		V ac
Number of DC distributions		
Voltage level of these		V dc

Number of battery-backed busbar systems		
Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)		
Number of turbine sections per unit (e.g. HP/LP/LP)		
Turbine speed		rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/°C

Generator

Type		
Rated power		MVA
Active power		MW
Voltage		kV
Frequency		Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type		
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate		m^3/s
Cooling water temperature		°C
Condenser pressure		hPa

Condensate pumps

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

Condensate cleanup system

Full flow/part flow		
Filter type		

Feedwater tank

Volume		m^3
Pressure/temperature		MPa/°C

Feedwater pumps

Number		
Flow rate		kg/s
Pump head		
Feedwater temperature		°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages		
Redundancies		

4.11.9 Project status and planned schedule

No information provided.

References

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[3] WWER-1000 description in "Review of design approaches of advanced pressurized LWRs." Report of a Technical Committee Meeting and Workshop held in Moscow, Russian Federation, 10-13 May 1994, IAEA-TECDOC-861, 1996.

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CHAPTER 5. EVOLUTIONARY, MEDIUM SIZE ALWR DESIGNS (the 500-800 MWe range)

5.1 AC-600, CHINA NATIONAL NUCLEAR CORPORATION, CHINA

5.1.1 Introduction

The design of the AC-600 (Advanced Chinese PWR) is based on the design of the Qinshan phase II standard PWR nuclear power plant (2×600 MWe) with the objective of improving the economy and safety of the nuclear power plant through use of system simplification, passive safety, and modular construction. The AC-600 is expected to become a major type of reactor for the next generation of 600 MWe nuclear power plants in China.

The AC-600 has a large safety margin of operation because of the small power density of the reactor core. The high natural circulation cooling ability due to a small flow resistance of the primary system loop is very useful for reactor core decay heat removal during accidents. The AC-600 has a large reactor pressure vessel, a large pressurizer and a large water volume in the primary systems so as to function as accident mitigation. The AC-600 design, eliminating the high head safety injection pumps, utilizes full-pressure core makeup tanks and larger accumulators for the engineered safety features. The passive containment cooling system is used as the ultimate heat sink. All the measures mentioned above increase both the reliability and the capacity of the engineered safety systems very much, largely improving the safety of AC-600. The major design targets of AC-600 are given in Table 5.1-I.

The safety goals of nuclear power plants should include not only the protection of the environment and the public, but also the protection of the plants themselves as well. The two sides of the safety goals can not be separated completely but are closely related to each other. It is quite evident that the goals of protecting the environment and public health can only be attained under the prerequisite of the safety of the nuclear power plants themselves. Increasing the plant's own safety and preventing core melt should be emphasized so as to restore the public confidence in nuclear power.

5.1.2 Description of the nuclear systems

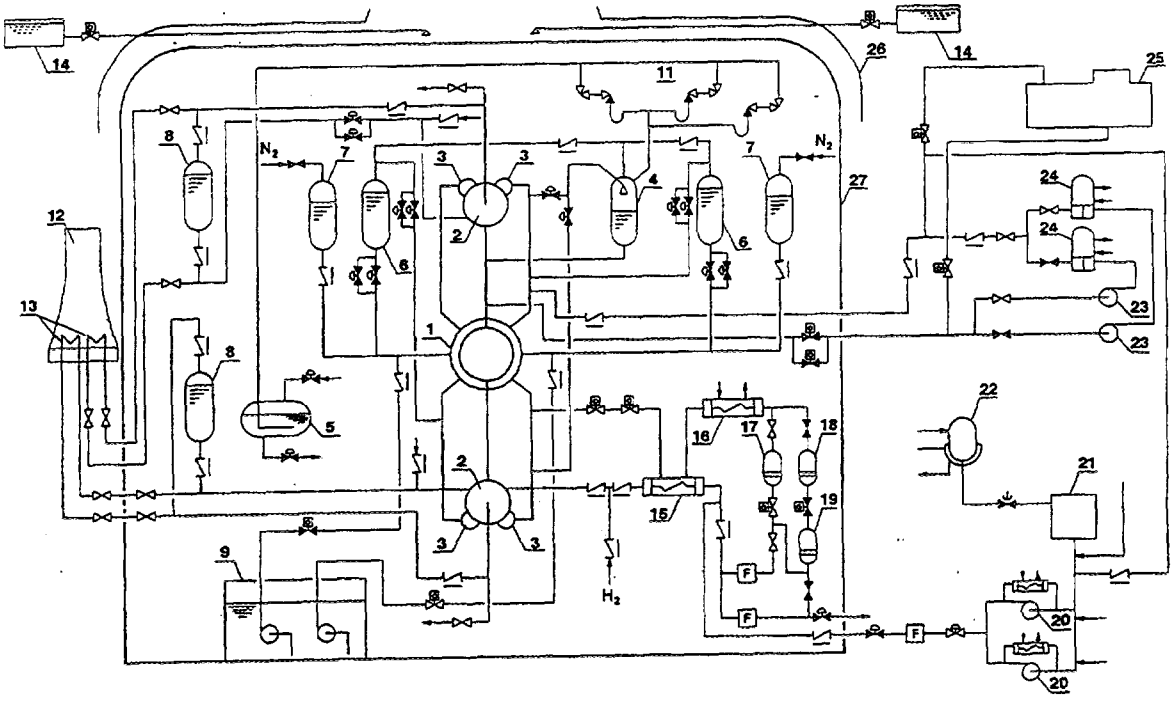
5.1.2.1 Primary circuit and its main characteristics

The AC-600 reactor plant design is based on the design of Qinshan phase II (2×600 MWe PWR NPPs), but it incorporates a number of improvements and safety enhancements compared with the plants of Qinshan phase II (QS-II).

The primary circuit of the AC-600 uses 2 loops with a steam generator and two reactor coolant pumps in a "one-hot-leg-two-cold-legs" arrangement connected in parallel and symmetrically to the reactor, a pressurizer, and a relief tank (Figure 5.1-1 and 5.1-2).

Table 5.1-I: Major design targets for the AC-600

Parameter	Design Target
Construction cost	about 20% less than that of Qinshan Phase II NPP
Core melt frequency	1×10^{-5} to 1.5×10^{-6}/r-y
Availability factor	> 85%
Refuelling period	18 months
Construction period	5 to 6 years
Plant life time	60 years
Plant personnel exposure dose	50-100 man-rem/year



Legend:

Item	Name	Quantity
1	reactor	1
2	steam generator	2
3	primary coolant pump	4
4	pressurizer	1
5	relief tank	1
6	core makeup tank	2
7	accumulator	2
8	emergency water tank	2
9	special sump	2

Item	Name	Quantity
10	low pressure safety injection pump	4
11	safety valve	3
12	chimney	1 or 2
13	emergency air cooler	2
14	water storage tank	1
15	regenerative heat exchanger	1
16	letdown heat exchanger	1
17	mixed bed exchanger	1
18	mixed bed exchanger	1

Item	Name	Quantity
19	cation bed exchanger	1
20	makeup water pump	2
21	boric acid storage tank	1
22	boric acid makeup tank	1
23	spent fuel pit cooling pump	2
24	spent fuel pit heat exchanger	2
25	spent fuel pit	1
26	protective shell	1
27	containment (steel)	1

Figure 5.1-1 AC-600 nuclear island, flow diagram

In the AC-600 design, the vertical distance between the steam generators and the reactor core has been increased, and the flow resistance of the coolant loops have been reduced, to increase the natural circulation cooling flow rate of the primary coolant. If the reactor operates at 25% of rated power, the natural circulation flow is 4852 t/h = 1347.78 kg/s (15.12% of the rated flow rate) after the reactor coolant pumps shut down. The natural circulation flow rate increase is a very important part of the passive safety characteristics of the AC-600.

5.1.2.2 Reactor core and fuel design

The reactor core consists of 145 17×17 advanced fuel assemblies, 57 control rod assemblies and other fuel associated assemblies. There are 45 black rod (Ag-In-Gd) and 12 grey rod (stainless steel) assemblies in the core.

The average linear power density of the AC-600 fuel rod is 13.78 kW/m, much smaller than that of Qinshan phase II (16.09 kW/m). The small core power density makes for the reactor to have large thermal safety margins for normal operation and accident conditions. The average burnup is 42 000 MWd/tU.

The AC-600 design uses Gd_2O_3 burnable poison, solidly melted in the fuel, to reduce the excess reactivity of the reactor and the critical boron concentration.

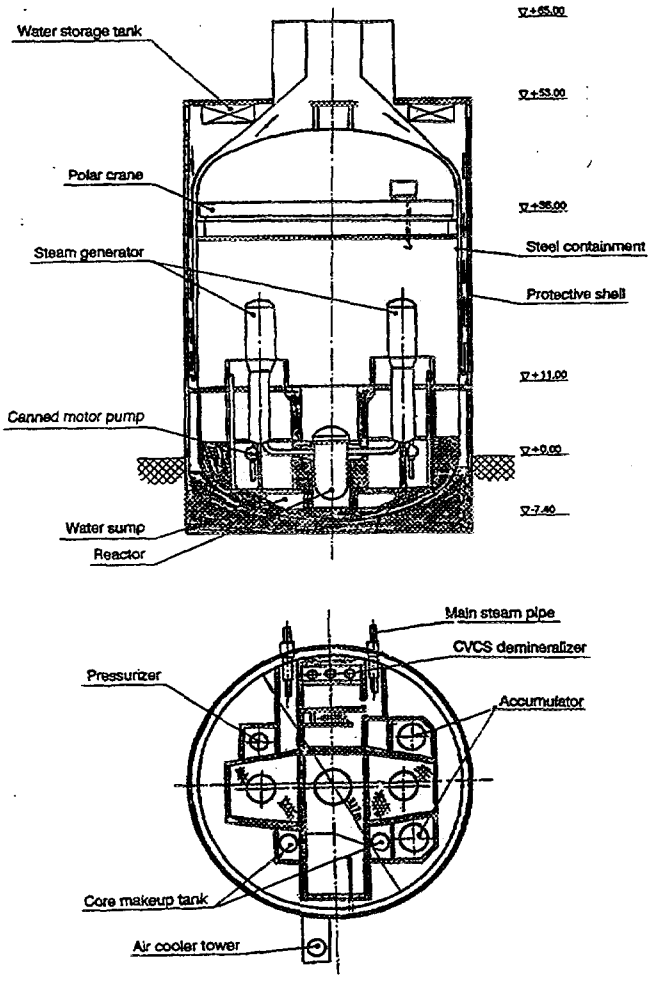


Figure 5.1-2 Reactor building arrangement

Due to the small critical boron concentration, a large negative temperature coefficient of reactivity can be obtained. The small excess reactivity and the large negative temperature coefficient of the core is one of the AC-600 design characteristics, largely improving the passive and inherent safety of the reactor to prevent power excursions induced by reactivity accidents.

The control rod drive mechanisms (CRDMs) for the Qinshan phase II plants will be adopted in the design for the AC-600, except that the wires to be used in the electromagnetic coils for the AC-600 CRDMs are melting-extruded. The operating temperature of the coils will be more than 300°C (about 350°C).

5.1.2.3 Fuel handling and transfer systems

No information provided.

5.1.2.4 Primary components

Reactor pressure vessel

The reactor vessel (Figure 5.1-3) encloses all components of the reactor core. It is made of SC 508-3 steel made in China. Due to the lower power density in the core and the larger vessel inside diameter, it is considered to be much safer during the 60 years design life of the plant. The nozzles of the control rod drive mechanisms (CRDM) and the in-core instrumentation are located on the closure head. There are no penetrations in the reactor pressure vessel below the level of the reactor coolant nozzles.

Reactor internals

No information provided.

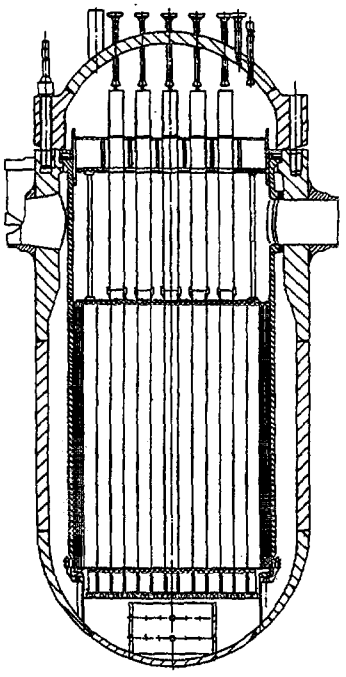


Figure 5.1-3 Reactor vessel and internals

Steam generators

The steam generators are of the vertical U-tube type used in Qinshan phase II. The material of the U-tubes is Inconel-690. Two canned pumps are welded reversely on the steam generator bottom head. In this case, the U shape cross-over leg of the coolant pipe is eliminated.

Pressurizer

The pressurizer is the same as for the Qinshan phase II. Its total volume is 36 m^3.

Reactor coolant pumps

The reactor coolant pumps (RCPs) are of the mixed flow, canned motor pump type. There are four canned pumps connected to the steam generator bottom heads directly. Lubrication and cooling of the RCPs are performed with water. To increase the rotating inertia of the canned motor pump, a motor and pump design with a rotating inertia of 0.15 t-m^2 will be employed.

Main coolant lines

The inside diameter of the hot and cold leg nozzles is 787.4 and 520.7 mm, respectively.

5.1.2.5 Reactor auxiliary systems

Chemical and volume control system

No information provided.

Emergency core cooling system

The AC-600 utilizes an emergency core cooling system that is based on the principle of combining passive and active features. There are 3 subsystems for the emergency core cooling system.

The high pressure injection subsystem consists of 2 reactor core makeup tanks. The middle pressure injection subsystem consists of 2 accumulators. The low pressure injection and long term cooling subsystem consists of 4 low pressure injection pumps taking suction from 2 special sumps in the containment. The low pressure injection pumps are of the vertical phreatic water type. The main functions of the emergency core cooling system are as follows:

- To supply water to the reactor in the event of abnormal leakage.
- In the event of LOCA, to inject water into the reactor core and provide long term core cooling.

Residual heat removal system

This is an entirely passive system which has the function of removing the reactor core residual heat when the normal cooling is lost, as a result of station blackout or other accidents.

This system comprises two trains. Each train consists of an emergency water tank and an emergency air cooler. When a station blackout or other accident occurs, the isolation valves located at the outlet pipe of the emergency water tank are opened by a low-low water level signal for the steam generator, so that the emergency water tanks provide water to the secondary side of the steam generator by gravity and maintain the water level. The water in the steam generator absorbs the heat from the reactor coolant, when the water is heated into steam. The steam rises and passes through the emergency air coolers where the steam condenses; simultaneously, the heat is transferred to the atmosphere. The condensed water flows back to the steam generators by gravity; thereby a continuous natural circulation path will be established. Because of the cooling of the secondary side of steam generators, the corresponding natural circulation in the reactor coolant system will also be established. In this way, the residual heat of the reactor core will be transferred to the atmosphere.

5.1.2.6 Operating characteristics

No information provided.

5.1.3 Description of turbine generator plant system

5.1.3.1 Turbine generator plant

No information provided.

5.1.3.2 Condensate and feedwater systems

No information provided.

5.1.3.3 Auxiliary systems

No information provided.

5.1.4 Instrumentation and control systems

5.1.4.1 Design concepts including control room

The monitoring and control system provides an automated diagnosis of the state and the operating conditions of the nuclear power plant. Monitoring and presentation of information on the reactor coolant system, on all the safety-related systems, on the containment, on all operating conditions of the plant and on remote control of these systems is provided. A post-accident monitoring system is provided to estimate the state of the plant.

Facilities for presentation of information including displays for monitoring of safety systems ensure:

- indication of control rod position
- monitoring of neutron flux during operation, refuelling and maintenance
- monitoring of level of radioactive contamination of the containment and the surrounded area
- preservation of adequate water level in the reactor vessel and the cooling systems
- scram of the reactor
- protection of safety-related systems.

In case of a main control room failure, the reserve control room is to provide:

- reactor trip to hot shutdown condition
- maintaining of hot shutdown condition

- monitoring of subcriticality
- putting into operation of confining systems
- reactor cooldown with some local operations.

Plant process control systems fulfil the automatic control of the following main controlled parameters:

- neutron flux in the core
- primary pressure
- secondary pressure
- water level in the steam generators
- water level in the pressurizer

The design value of the reactor neutron flux is maintained within $\pm 2\%$ of its nominal value by means of the control bank of neutron absorbers, consisting of several rod cluster control assemblies,.

The design value of the primary pressure is maintained within ± 0.3 MPa by means of the pressurizer electric heaters and by spray valves on the pressurizer spray line from the reactor coolant pump exit side to the steam phase of the pressurizer.

The design value of the secondary pressure is maintained within ± 0.2 MPa by an appropriate balance of reactor power and steam flow from the steam generators to the turbine or to the steam dumping devices.

The design value of the water level in the steam generators is maintained within ± 180 mm of its nominal level by means of the steam generator feedwater supply controller, actuating the control valve on the steam generator feedwater line.

The design value of the water level in the pressurizer is provided by the level controller, actuating the control valves located on the make-up line, and make-up pumps, to keep within normal value.

During normal operation, the reactor's neutron power and the process parameters are maintained automatically by the reactor control system. Protection against transients due to the introduction of reactivity is ensured by the reactor protection system. When reaching the setpoints of neutron flux or reactor period, the reactor protection system will warn the operators to take actions or will trip the reactor so that reactor safety can be ensured.

5.1.4.2 Reactor protection system and other safety systems

The degree of automation is such that reactor safety will be ensured fully by automatic control and protection systems during the first 30 minutes after an accident has happened. In design basis accident situations, interventions by the operators should not take place within the first 30 minutes in order to allow the operators enough time to consider the characteristics of the accident, which is seen as an effective countermeasure to prevent erroneous actions. The reactor will be "walk-away" safe during this period.

In addition, in design basis accident situations provisions are made for:

- accident state monitoring, such as in-core and sump level monitoring
- indication of control rod position, including lights and digits
- indication of radiation level and radioactive releases
- monitoring of the reactor safety shutdown states

These systems are provided with devices for automatic event recording during accidents. Alarm light signals and digital indications are also provided in the central control room.

The list of automatic safety systems encompasses:

- reactor scram system;
- primary overpressure protection system;

- emergency core cooling system;
- system of passive residual heat removal from the secondary side of the steam generator;
- passive cooling system for the containment;
- system of quick-acting isolation valves in the main steam lines;
- secondary overpressure protection system;
- diesel-generator system and
- system of reliable direct current power supply.

Except for the two last systems, these systems are described in Section 5.1.6.2; for the remaining two reference is made to Section 5.1.5.2.

When an accident occurs, the reactor protection system is designed to initiate a reactor trip, scram, which is actuated by de-energizing the control rod drive mechanisms. The following parameters are used to trigger the reactor protection system:

- increase of neutron flux
- decrease of reactor period
- OTΔT and OPΔT
- decrease of pressure in the reactor
- increase of pressure in the reactor
- decrease of the flow rate in the reactor
- decrease of water level in SG
- increase of water level in SG
- decrease of the reactor coolant pump speed
- signal of safety injection

The reactor protection system is designed to trip the reactor when the parameters go beyond preset limits, to ensure compliance with the design criteria under all design conditions.

In order to mitigate the consequences of a failure to scram (ATWS = Anticipated Transients Without Scram), the protection actions will include tripping of the turbine and start of the auxiliary feedwater system.

5.1.5 Electrical systems

5.1.5.1 Operational power supply systems

The normal and the emergency electric power supply system consists of two trains of 100% capacity, with each channel being divided into three groups considering reliability requirements and the time interval of loss of electric power.

5.1.5.2 Safety-related systems

Diesel generator system

Two physically separated diesel generators provide power supply to the safety-related systems, involving the recirculation pumps of the subsystem of low pressure active recirculation.

Start-up of the two diesel-generators, one for each channel of reliable electric power and to be put into operation in the case of failure of main and reserve grid connections, is carried out for a period not exceeding 15 s from the moment of a command to start-up.

System of reliable direct current power supply

This system consists of storage batteries. It provides the power supply to electromagnetic circuits for operating of safety systems and for recording of necessary post-accident parameters.

D.C. electric power supply of the reactor control and protection system is ensured by batteries (in each train) designed for a discharge over 24 hours. Electric power supply from accumulator batteries during

a station blackout situation is provided for the main control room and the auxiliary control room in full measure.

5.1.6 Safety concept

5.1.6.1 Safety requirements and design philosophy

The basic safety requirements can be summarized as:

- During normal conditions and operational occurrences, radioactive material shall be confined by maintaining the integrity of all physical barriers in the defence-in-depth: fuel rod claddings, primary pressure boundary and steel containment.
- Radioactive materials released from the primary circuit in design basis accidents shall be confined by maintaining steel containment integrity.
- In the event of design basis accidents with a leak from primary to the secondary circuit the release of radioactive material shall be controlled by isolation of the steam generators on the steam and water side by means of quick-acting shut off isolation valves, actuated by a signal of radioactivity increase in the damaged steam generator.
- Radioactive materials released from the fuel and the primary circuit in beyond-design accidents shall be confined by the steel containment and by operation, if necessary, of the filtration plant for controlled removal of the atmosphere inside the containment.

In order to meet these requirements, the AC-600 design is based on utilization of a combination of active and passive features. Quite obviously, controls and countermeasures related to normal operation and occurrences rely on active systems and components in the same way as currently operating PWRs, but when it comes to accident situations, the strategy involves using passive safety features, as far as possible and practible for short term actions and responses, whereas active systems are provided as back-up and means for corrective and mitigative actions in the longer term.

The essential safety functions in emergency situations are to shut the reactor down, and to remove the residual heat. Shutdown is accomplished by a reliable system that acts in a passive way after initiation, as described below, and residual heat is removed by means of completely passive system functions. The safety injection system which serves to provide make-up water in the event of leakages, comprises a set of three systems, the two systems for high and medium pressure range injection are passive, whereas the third, for low pressure injection, is an active system.

In the first stage of a normal plant shutdown, the residual heat of the reactor and the coolant system is transferred to the secondary loop through the steam generators. The steam then generated is through the turbine bypass system conveyed to a condenser to be condensed. The auxiliary feedwater system supplies the steam generator with water. The whole process goes on till the pressure of the coolant system has decreased to 2.8 MPa and the temperature to 180°C.

In the second stage of the shutdown, residual heat removal is accomplished by the residual heat removal system. The residual heat removal system and the spent fuel pool cooling and purification system share the same equipment. It consists of two independent trains, each of which includes one pump and a heat exchanger cooled by equipment coolant. During normal plant operation, this system serves as the spent fuel pool cooling and purification system. During plant shutdown, one of its trains is used for reactor residual heat removal. At the same time, the spent fuel pool is also cooled till the coolant pressure is below 0.1 MPa. Coolant temperature decreases to and remains at cold shutdown temperature.

During the plant shutdown and cooling process, the coolant pumps are always in operation; they do not stop until the coolant temperature has decreased to 70°C. During this period, reactor coolant is circulated through the coolant loops. After that, the coolant is circulated by the spent fuel pit cooling pump.

In the event of a large leakage from the reactor coolant pressure boundary, the residual heat is removed by the safety injection or emergency core cooling system and the passive containment cooling system. Make-up water to the primary system to ensure that the fuel assemblies in the pressure vessel will remain covered with water is provided by the emergency core cooling system.

As long as the reactor coolant pressure boundary is completely intact, a station blackout is a typical situation in which emergency residual heat removal will be required. In such case, the passive heat removal system on the secondary side of the steam generators is automatically put into operation. Through natural circulation of primary coolant, natural circulation of secondary loop steam and condensed water, and natural convection of air in special ducts outside the containment, residual heat is removed to the atmosphere. By this system, the coolant temperature and pressure can be kept within acceptable limits until the power supply has been restored, or brought down to values corresponding to cold shutdown. Besides the condensed water from the air cooled heat exchanger, secondary side system feedwater is also available from the emergency feedwater tank. So, the water volume is kept at the required value by natural circulation in the secondary system.

Deterministic design basis

No information provided.

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

The major safety systems with respect to shutdown, core cooling, and overpressure protection are:

- Reactor scram system
- Primary circuit overpressure protection system
- Emergency core cooling system
- System of passive residual heat removal from the secondary side of the steam generator
- Passive cooling system for the containment
- System of quick-acting isolation valves in steam lines
- Secondary circuit overpressure protection system

which are described further in the following.

Reactor scram system

The reactor scram system provides reliable switch-off of the electric power supply to the rod drive, causing the emergency shutdown rods to drop into the core. In this case, the disappearance of the signal of the original cause does not stop the initial action of the emergency protection.

Primary circuit overpressure protection system

The system comprises three identical pilot safety valve assemblies, which discharge steam or steam-water mixture from the steam phase of the pressurizer to the relief tank when the pressure in the pressurizer increases above the permissible one. The subsystem for receiving the steam or steam-water mixture involves a relief tank and pipelines connecting it with the outlets of the safety valves.

Emergency core cooling system

The emergency core cooling system (ECCS) comprises the following complex of subsystems initiated automatically:

- system of core make-up tank with full pressure (high pressure safety injection subsystem)
- system of accumulator with nitrogen under pressure
- system of low pressure active safety injection and recirculation

In order to increase the reliability of the safety injection, two full pressure core makeup tanks, two accumulators and four low head safety injection/recirculation pumps, which are installed in the containment sumps, are utilized in the AC-600 design. AC power supplies are not needed for fulfillment of the ECCS

functions, except for the subsystems of low pressure active safety injection and recirculation. The air-operated valves, needed for the function of emergency heat removal, are driven by compressed air from compressed air storage tanks.

In case of a large LOCA, the flow rate into the RCS from a core makeup tank is larger than that from a high head safety injection pump in a conventional design. In order to ensure the functions of the low head safety injection/recirculation system, utilization of active pumps has been found necessary. The power supply of the subsystems of low pressure active safety injection and recirculation are provided by the diesel generators or by the offsite power source (during the recirculation stage after LOCA).

System of passive residual heat removal from the secondary side of the steam generator

The passive emergency residual heat removal system is mainly used to remove the residual reactor power in the event of a station blackout by natural circulation in the primary coolant system and on the secondary side of the steam generator. It may also be effective in a main steam line rupture or loss of feedwater event. The system consists of two independent trains, each of them being connected to the reactor coolant loops via the respective steam generator. Each train has an emergency feedwater tank, an emergency air cooler located outside the containment, and piping (and valves) for steam and condensate circulation. The fail-open valves on the piping are driven by compressed air. The air-cooler rejects the core decay heat transferred from the steam generators into the atmosphere outside the containment.

Passive cooling system for the containment

The passive containment cooling system is used to remove the heat from the inside to the outside of the containment during a LOCA or a main steam line rupture inside the containment. The steam released is condensed on the inside of the containment shell which is cooled on the outside by a natural circulation air flow and a gravity-driven water spray onto the steel shell surface by water from the elevated tanks on top of the containment. The heat released inside the containment is rejected to the atmosphere from the containment, and the pressure and temperature inside the containment decrease. The pressure of the atmosphere inside the containment is kept below the permissible design value.

After the tank on the top of the containment has become empty, the natural circulation flow of air through the annulus between the steel shell and the concrete shell can remove the heat from the inside to the outside of the containment continuously. At the same time, the low head safety injection/recirculation pumps which are installed in the containment sumps can withdraw the borated water from the sumps into the reactor coolant system. The water absorbs the core decay heat and flows out through the break point (in LOCA situations).

System of quick-acting isolation valves in steam lines

The quick-acting isolation valves in the steam lines close at:

- water inventory in steam generators increases above the permissible level;
- increase of radioactivity in steam generators above the permissible level; and
- reception of signals of a steam line rupture.

The system provides for

- protection of the turbine from steam of high humidity;
- prevention of radioactivity release from steam generators; and
- restriction of steam blowdown during rupture of the secondary circuit.

Secondary circuit overpressure protection system

This system prevents the secondary circuit pressure to increase above the permissible level of 110% of secondary design pressure. It incorporates a power operated relief valve and seven safety valves. These valves reject steam into the atmosphere.

5.1.6.3 Severe accidents (Beyond design basis accidents)

In order to prevent the core from melting or radioactive release from the plant to the environment, operators are required to utilize all reasonable measures according to procedure H or U.

The fission chain reaction in the core should be stopped during severe accidents and the reactor returned to a controllable state. The measures of mitigation and accident management are researched through use of severe accident analysis.

5.1.7 Plant layout

5.1.7.1 Buildings and structures, including plot plan

Design bases

The design of the AC-600 is taking into account two levels of seismic events: an operation basis earthquake (OBE) of magnitude 7 on the MSK-64 scale and a safe shutdown earthquake (SSE) of magnitude 8 on the MSK-64 scale.

The reactor plant equipment is calculated for seismic effects. During the operating basis earthquake, normal operation of the reactor plant is to be provided. During the safe shutdown earthquake reactor, a safe plant shutdown and adequate cooling of the core must be ensured. All civil structures, process components and equipment, pipelines, instrumentations, and so on, depending upon the degree of their responsibility for safety ensurance during seismic events and availability after an earthquake, are divided into 3 seismic categories. Components and systems of category 1 shall fulfill their safety functions during and after an earthquake of SSE intensity. After an OBE, availability is maintained.

The seismic category 1 includes:

- Systems for normal operation, failure of which during an SSE may results in radioactivity releases causing excessive population doses in comparison with the specified values for SSE condition.
- Safety systems for keeping the reactor in a subcritical state, for emergency heat removal and for confinement of radioactive products.
- Structures and equipment which could impair these functions as a consequence of an SSE.

The designers are considering the possibility of using special seismic isolators located under the base plate to minimize the seismic impact on structures and equipment.

The external wind load for the first category buildings and structures is assumed to arise from a hurricane with wind speed of 25 m/s.

With respect to tornado effects, the following characteristics and physical parameters are taken into account in the design for the first category buildings and constructions:

- Maximum horizontal speed of rotation of tornado wall: 85 m/s.
- Translational motion speed of tornado: 22 m/s.
- Tornado radius: 45 m.
- Differential pressure between centre and periphery of the whirlwind: 8.5 kPa.
- Impact of missiles carried away by a whirlwind with a speed of 26 m/s.

With respect to external explosions and airplane crashes, the following characteristics are specified:

- Front pressure of explosion shock wave: 50 kPa.
- Duration of compression phase: 300 ms.
- Direction of propagation is horizontal.
- Impact of a plane with 5.7 t mass at a speed of 100 m/s.

The size and energy of missiles inside the steel containment are determined in the design based on the "leak before break" concept. The mechanical effect of missiles and steam-water jets on the steel containment is excluded by means of a protective shield.

The steel containment is designed to withstand an internal overpressure of 0.4 MPa and a maximum temperature of 134°C with a very high leaktightness; the leakage shall not exceed 0.3% of volume per day. During design basis accidents, the containment systems ensure confinement of radioactive material inside the steel containment, heat removal from the hermetic steel containment, and control and suppression of hydrogen.

Physical separation aspects, & Radiation protection aspects (accessibility, shielding, ventilation)

No information provided.

Plot plan

No information provided.

5.1.7.2 Reactor building

No information provided.

5.1.7.3 Containment

Containment integrity

No information provided.

Corium retention

No information provided.

5.1.7.4 Turbine building

No information provided.

5.1.7.5 Other buildings

Safeguard and fuel buildings

No information provided.

5.1.8 Technical data

General plant data		
Power plant output, gross	600	MWe
Power plant output, net		MWe
Reactor thermal output	1 936	MWt
Power plant efficiency, net		%
Cooling water temperature		°C
Nuclear steam supply system		
Number of coolant loops	2	
Primary circuit volume, including pressuriser		m^3
Steam flow rate at nominal conditions [1951t/h]	2168	kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	-/6.65	°C/MPa
Feedwater temperature/pressure	230/	°C/MPa
Reactor coolant system		
Primary coolant flow rate [13.194m^3/s]	9 780	kg/s
Reactor operating pressure	15.8	MPa
Coolant inlet temperature, at RPV inlet	293	°C
Coolant outlet temperature, at RPV outlet	327	°C
Mean temperature rise across core		°C
Reactor core		
Active core height	3.66	m
Equivalent core diameter	2.92	m
Heat transfer surface in the core	6 222.7	m^2
Total fuel inventory [UO_2]	66.8	t
Average linear heat rate	13.42	kW/m
Average fuel power density	32.9	kW/kg U
Average core power density (volumetric)	78.69	kW/l
Thermal heat flux, F_q	311.1	kW/m^2
Enthalpy rise, F_H		

Fuel material	Sintered UO_2
Fuel assembly total length	4 100 mm
Rod array	square, 17×17 -25 (AFA)
Number of fuel assemblies	145
Number of fuel rods/assembly	264
Number of guide tubes for control rods/instr.	24/1
Number of spacers	
Enrichment (range) of first core	1.9, 2.5, 3.1 Wt%
Enrichment of reload fuel at equilibrium core	3.4 Wt%
Operating cycle length (fuel cycle length)	18 months
Average discharge burnup of fuel	42 000 MWd/t
Cladding tube material	Zr 4
Cladding tube wall thickness	0.57 or 0.64 mm
Outer diameter of fuel rods	9.5 mm
Overall weight of assembly	kg
Active length of fuel rods	3 660 mm
Burnable absorber, strategy/material	Gd_2O_3, mixed with fuel
Number of control rod assemblies	57 (45 black & 12 gray rods)
Absorber rods per control assembly	20
Absorber material, black/gray rods	Ag-In-Cd / stainless steel
Drive mechanism	Magnetic jack
Positioning rate	mm/min (or mm/s)
Soluble neutron absorber	
Reactor pressure vessel	
Cylindrical shell inner diameter	4 000 mm
Wall thickness of cylindrical shell	205 mm
Total height	12 220 mm
Base material: cylindrical shell	low alloy steel A508-III
RPV head	
Design pressure/temperature	17.2/343 MPa/°C
Transport weight (lower part) [include. head]	390 t
RPV head	t

Steam generators

Type	Vertical, U-tube heat exchanger	
Number	2	
Heat transfer surface	5 430	m^2
Number of heat exchanger tubes	4 474	
Tube dimensions	19.05x1.09	mm
Maximum outer diameter	3 496	mm
Total height	21 000	mm
Transport weight	350	t
Shell and tube sheet material	A508-III	
Tube material	Inconel 690	

Reactor coolant pump

Type	Single-stage centrifugal pump with canned motor	
Number 4	4	
Design pressure/temperature	17.2/343	MPa/°C
Design flow rate (at operating conditions){3.299m^3/s]	2 445	kg/s
Pump head		
Power demand at coupling, cold/hot	3 340/2 545	kW
Pump casing material	Stainless steel	
Pump speed	1 500	rpm

Pressuriser

Total volume	36	m^3
Steam volume: full power/zero power	14.4/23.6	m^3
Design pressure/temperature	17.2/360	MPa/°C
Heating power of the heater rods	1 440	kW
Number of heater rods	60	
Inner diameter	2 100	mm
Total height	11 000	mm
Material	A508=III	
Transport weight	90	t

Pressuriser relief tank

Total volume	m^3
Design pressure/temperature	MPa/°C
Inner diameter (vessel)	mm
Total height	mm
Material	
Transport weight	t

Primary containment

Type	Dry, double wall, in steel/concrete	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	37/57	m
Free volume	75 000	m^3
Design pressure/temperature (DBEs)	430/	kPa/°C
(severe accident situations)	-/-	kPa/°C
Design leakage rate	< 0.25	vol%/day
Is secondary containment provided?	yes, space between the walls	

Reactor auxiliary systems

Reactor water cleanup,	capacity	kg/s
	filter type	
Residual heat removal,	at high pressure	kg/s
	at low pressure	kg/s
Coolant injection,	at high pressure	kg/s
	at low pressure	kg/s

Power supply systems

Main transformer,	rated voltage	kV
	rated capacity	MVA
Plant transformers,	rated voltage	kV
	rated capacity	MVA
Start-up transformer	rated voltage	kV
	rated capacity	MVA
Medium voltage busbars (6 kV or 10 kV)		
Number of low voltage busbar systems		
Standby diesel generating units: number		
	rated power	MW
Number of diesel-backed busbar systems		
Voltage level of these		V ac
Number of DC distributions		
Voltage level of these		V dc

Number of battery-backed busbar systems		
Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)		
Number of turbine sections per unit (e.g. HP/LP/LP)		
Turbine speed		rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/°C

Generator

Type		
Rated power		MVA
Active power		MW
Voltage		kV
Frequency		Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type		
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate		m^3/s
Cooling water temperature		°C
Condenser pressure		hPa

Condensate pumps

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

Condensate clean-up system

Full flow/part flow		
Filter type		

Feedwater tank

Volume		m^3
Pressure/temperature		MPa/ °C

Feedwater pumps

Number		
Flow rate		kg/s
Pump head		
Feedwater temperature		°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages		
Redundancies		

5.1.9 Project status and planned schedule

No information provided.

References

No information provided.

5.2 AP-600, WESTINGHOUSE

5.2.1 Introduction

The Westinghouse Advanced Passive PWR AP-600 is a 600 MWe pressurized water reactor (PWR) with advanced passive safety features and extensive plant simplifications to enhance the construction, operation, and maintenance of the plant. The plant design utilizes proven technology which builds on over 30 years of operating PWR experience. PWRs represent 76 percent of all Light Water Reactors around the world, and 67 percent of the PWRs are based on Westinghouse PWR technology.

The AP-600 is designed to achieve a high safety and performance record. It is conservatively based on proven PWR technology, but with an emphasis on safety features that rely on natural forces. Safety systems maximize the use of natural driving forces such as pressurized gas, gravity flow and natural circulation flow. Safety systems do not use active components (such as pumps, fans or diesel generators) and are designed to function without safety-grade support systems (such as AC power, component cooling water, service water, HVAC). The number and complexity of operator actions required to control the safety systems are minimized; the approach is to eliminate a required operator action rather than to automate it. The net result is a design with significantly reduced complexity and improved operability.

The AP-600 standard design complies with all applicable US NRC criteria. Extensive safety analysis has been completed and documented in the Standard Safety Analysis Report (SSAR) and Probabilistic Risk Analysis (PRA) submittals to the NRC. An extensive testing programme has been completed, and verifies that the innovative plant features will perform as designed and analyzed. PRA results show a very low core damage frequency which meets the goals established for advanced reactor designs and a low frequency of release due to improved containment isolation and cooling.

An important aspect of the AP-600 design philosophy focuses on plant operability and maintainability. These factors have been incorporated into the design process.

The AP-600 design includes features such as simplified system design to improve operability while reducing the number of components and associated maintenance requirements. In particular, simplified safety systems reduce surveillance requirements by enabling significantly simplified technical specifications.

Selection of proven components has been emphasized to ensure a high degree of reliability with a low maintenance requirement. Component standardization reduces spare parts, minimizes maintenance training requirements, and allows shorter maintenance durations. Built-in testing capability is provided for critical components.

Plant layout ensures adequate access for inspection and maintenance. Laydown space for staging of equipment and personnel, equipment removal paths, and space to accommodate remotely operated service equipment and mobile units have been considered as part of the plant design. Access platforms and lifting devices are provided at key locations, as are service provisions such as electrical power, demineralized water, breathing and service air, ventilation and lighting.

The AP-600 design also incorporates radiation exposure reduction principles to keep worker dose as low as reasonably achievable (ALARA). Exposure length, distance, shielding and source reduction are fundamental criteria that are incorporated into the design.

Various features have been incorporated in the design to minimize construction time and total cost by eliminating components and reducing bulk quantities and building volumes. Some of these features include the following:

- The flat, common basemat design selected for the nuclear island effectively minimizes construction cost and schedule.

- Utilization of the integrated protection system, the advanced control room, distributed logic cabinets, multiplexing, and fiber optics, significantly reduces the quantity of cables, cable trays, and conduits.
- A key feature of the AP-600 plant configuration is the stacked arrangement of the Class 1E battery rooms, the dc switchgear rooms, the integrated protection system rooms, and the main control room. This stacked arrangement eliminates the need for the upper and lower cable spreading rooms that are required in the current generation of PWR plants.
- Application of the passive safeguards systems replaces and/or eliminates many of the conventional mechanical safeguards systems that are typically located in the Seismic Category I buildings in the current generation of PWR plants.

The AP-600 is designed with environmental consideration as a priority. The safety of the public, the power plant workers, and the impact to the environment have all been addressed as specific design goals, as follows:

- Operational releases have been minimized by design features.
- Aggressive goals for worker radiation exposure have been set and satisfied.
- Total radwaste volumes have been minimized.
- Other hazardous waste (non-radioactive) have been minimized.

The AP-600 Nuclear Power Plant has been designed by Westinghouse under the sponsorship of the US Department of Energy (DOE) and the Electric Power Research Institute (EPRI). The design team includes a number of US and foreign companies and organizations, such as Bechtel, Burns & Roe, Initec (Spain), UTE (Spain), and Ansaldo (Italy) as architect engineers, Avondale Industries (module design), CBI Services, Inc. (containment vessel design), M-K Ferguson Co. (constructability, schedule, and cost estimation), Southern Electric International (turbine island buildings and systems), ENEA Energy Research Center of Italy (tests of the automatic depressurization system), SIET, SPES Facility in Italy (full-pressure integral passive safety system tests), and Oregon State University (low-pressure integral passive safety system tests).

The Electric Power Research Institute (EPRI) has, with a broad participation of numerous countries, developed a Utility Requirements Document (URD) for ALWRs, taking into account the wealth of information related to nuclear power plant safety and operations that has been generated worldwide with commercial nuclear power. The purpose of the URD is to delineate utility desires for their next generation of nuclear plants, and to this end, it consists of a comprehensive set of design requirements for future plants.

Incorporation of the ALWR URD has been a design goal for the AP-600 from the design inception, and has continued to be so during the ongoing First-of-a-Kind Engineering (FOAKE) program. The AP-600 has a well-defined design basis that is confirmed through thorough engineering analyses and testing and is in conformance with the URD. Some of the high-level design characteristics of the plant are:

- Net electrical power of at least 600 MWe; and a thermal power of 1940 MWt.
- Rated performance is achieved with up to 10% of the steam generator tubes plugged and with a maximum hot leg temperature of 600°F (315.6°C).
- Core design is robust with at least a 15% operating margin on core power parameters.
- Short lead time (five years from owner's commitment to commercial operation) and construction schedule (3 years).
- No plant prototype is needed since proven power generating system components are used.
- Major safety systems are passive; they require no operator action for 72 hours after an accident, and maintain core and containment cooling for a protracted time without ac power.
- Predicted core damage frequency of 1.7E-07/yr is well below the 1E-05/yr requirement, and frequency of significant release of 1E-08/yr is well below the 1E-06/yr requirement.
- Standard design is applicable to anticipated US sites.
- Occupational radiation exposure expected to be below 0.7 man-Sv/yr (70 man-rem/yr).
- Core is designed for a 24-month fuel cycle assuming an 87% capacity factor; capable of an 18-month cycle.
- Refuelling outages can be conducted in 17 days or less.

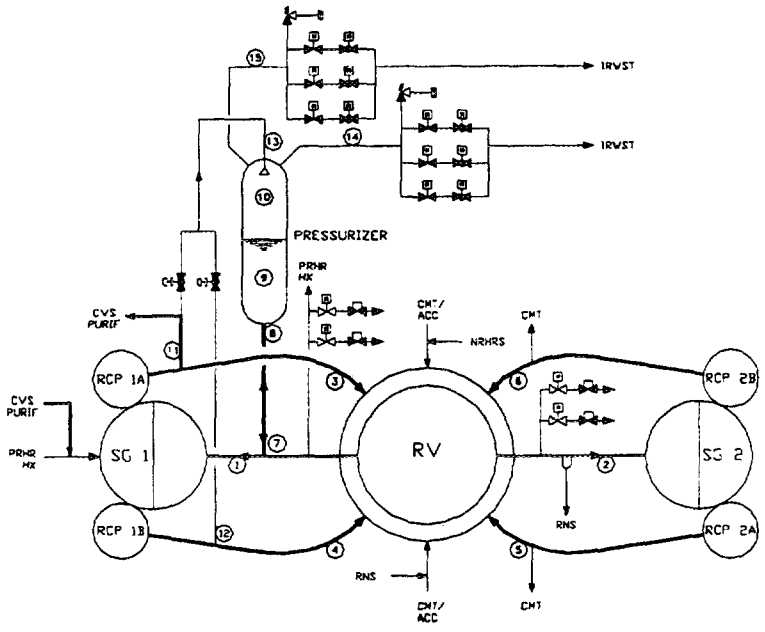


Figure 5.2-1 Reactor coolant system flow diagram

- Plant design life of 60 years without replacement of the reactor vessel.
- Overall plant availability greater than 90%, including forced and planned outages; the goal for unplanned reactor trips is less than one per year.

5.2.2 Description of the nuclear systems

5.2.2.1 Primary circuit and its main characterisitcs

The primary circuit of the AP-600 reactor has retained most of the general design features of current designs, but some evolutionary features that enhance the safety and maintainability of the system have been adopted. The coolant loops (Figure 5.2-2 and 5.2-1) consist of two hot leg and four cold leg pipes, and the reactor coolant pumps are installed directly onto the steam generators, eliminating the primary piping between pumps and steam generator; these features significantly contribute to safety and maintainability. Also, a simplified support structure for the primary systems reduces in-service inspections and improves accessibility for maintenance.

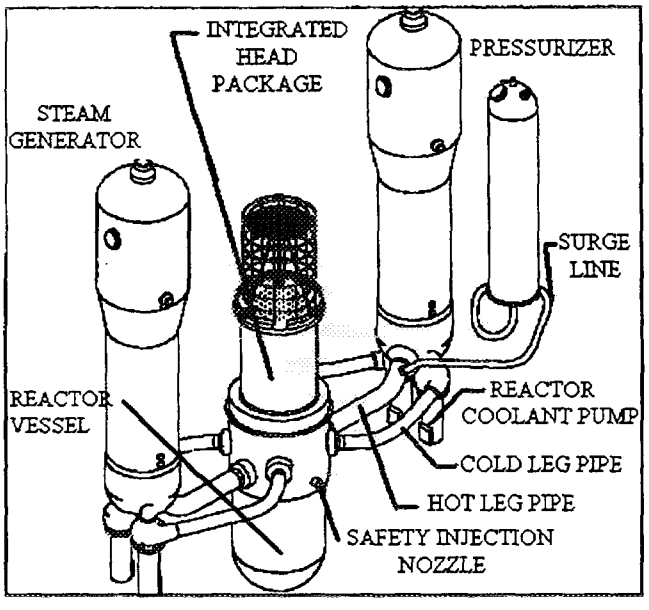


Figure 5.2-2 Isometric view of NSSS

The reactor coolant system pressure boundary provides a barrier against the release of radioactivity generated within the reactor and is designed to provide a high degree of integrity throughout operation of the plant.

5.2.2.2 Reactor core and fuel design

The core, reactor vessel, and reactor internals of the AP-600 are similar to those of a conventional Westinghouse PWR design. Several important features based on existing technology measurably enhance performance characteristics as compared with conventional plants. The reactor core is a low-power density design that uses a 12 foot (305 mm), 17×17 fuel assembly. Low-power density is achieved by making the core larger than previous 600 MWe designs, with the number of fuel assemblies increased from 121 to 145. This configuration results in core power density and average linear power density reductions of about 25 percent over conventional plants of the same power rating. This results in lower fuel enrichments, less reliance on burnable absorbers, and longer fuel cycles.

Another feature that contributes to lowering fuel cycle cost and extending reactor life is the use of a stainless steel radial neutron reflector. This reflector reduces neutron leakage, thereby improving core neutron utilization and allowing for reduced fuel enrichment. The radial reflector has the added benefit of reducing the damaging neutron fluence on the reactor vessel, an important factor in achieving the 60-year design life objective.

The combination of the radial reflector and the low-power density core results in a fuel-cycle cost savings of 15 to 20 percent compared with a standard design of the same power rating.

Another core design feature is the use of reduced-worth control rods (termed "gray" rods) to achieve daily load follow without requiring daily changes in the soluble boron concentration. The use of gray rods, in conjunction with an automated load follow control strategy, eliminates the need for processing thousands of gallons of water per day to change the soluble boron concentration sufficiently to achieve a daily load follow schedule. As a result, systems are simplified through the elimination of the evaporator, and other boron processing equipment (such as pumps, valves, and piping). With the exception of the neutron absorber materials used, the design of the gray rod assembly is identical to that of a normal control rod assembly.

The core consists of three radial regions that have different enrichments; the enrichment of the fuel ranges from 2 to 4%. The temperature coefficient of reactivity of the core is highly negative. The core is designed for a fuel cycle of 18 to 24 months with discharge burnups as high as 55 000 MWd/t.

5.2.2.3 Fuel handling and transfer systems

Refuelling of the reactor is performed in the same way as for current plants. After removing the vessel head, fuel handling takes place from above, using the refuelling machine. During refuelling, one third of the core inventory is replaced.

New fuel storage

New fuel is stored in a high density rack which includes integral neutron absorbing material to maintain the required degree of subcriticality. The rack is designed to store fuel of the maximum design basis enrichment. The rack in the new fuel pit consists of an array of cells interconnected to each other at several elevations and to supporting grid structures at the top and bottom elevations. The new fuel rack includes storage locations for 56 fuel assemblies. Minimum separation between adjacent fuel assemblies is sufficient to maintain a subcritical array even in the event the building is flooded with unborated water or fire extinguishing aerosols or during any design basis event.

Spent fuel storage

Spent fuel is stored in high density racks which include integral neutron absorbing material to maintain the required degree of subcriticality. The racks are designed to store fuel of the maximum design basis enrichment. Each rack in the spent fuel pool consists of an array of cells interconnected to each other

at several elevations and to supporting grid structures at the top and bottom elevations. The spent fuel storage racks include storage locations for 619 fuel assemblies. The modified 10×7 rack module additionally contains integral storage locations for five defective fuel storage containers. The design of the rack is such that a fuel assembly can not be inserted into a location other than a location designed to receive an assembly.

5.2.2.4 Primary components

Reactor pressure vessel

The reactor vessel (Figure 5.2-3) is the high pressure containment boundary used to support and enclose the reactor core. The vessel is cylindrical, with a hemispherical bottom head and removable flanged hemispherical upper head

The reactor vessel is approximately 38 feet (11.7 m) long and has an inner diameter at the core region of 157 inches (3.988 m). The total weight of the vessel is approximately 400 tons. Surfaces which can become wetted during operation and refuelling are clad with stainless steel welded overlay. The AP-600 reactor vessel is designed to withstand the design environment of 2500 psi (17.1 MPa) and 650°F (343 °C) for 60 years. The major factor affecting vessel life is radiation degradation of the lower shell.

As a safety enhancement, there are no penetrations below the top of the core. This eliminates the possibility of a loss of coolant accident by leakage from the reactor vessel which could allow the core to be uncovered. The core is positioned as low as possible in the vessel to limit reflood time in accident situations.

Reactor internals

The reactor internals, the core support structures, the core shroud, the downcomer and flow guiding structure arrangement, and the above-core equipment and structures, are very similar to those in current plants.

The reactor internals consist of two major assemblies - the lower internals and the upper internals. The reactor internals provide the protection, alignment and support for the core, control rods, and gray rods to provide safe and reliable reactor operation.

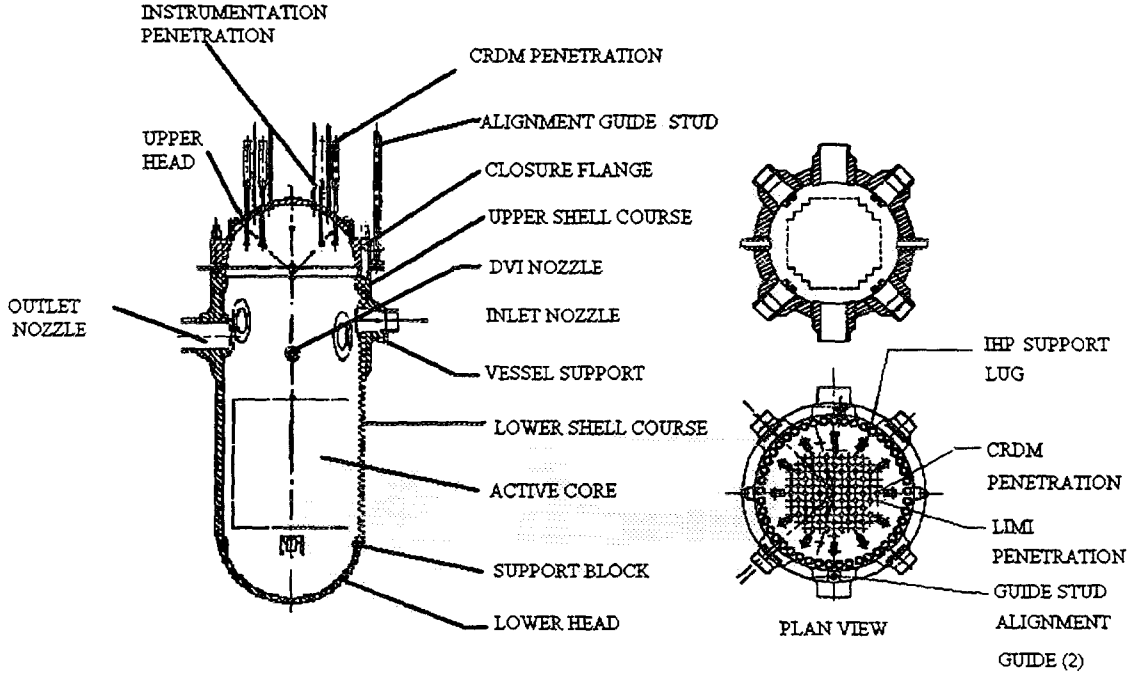


Figure 5.2-3 Reactor pressure vessel

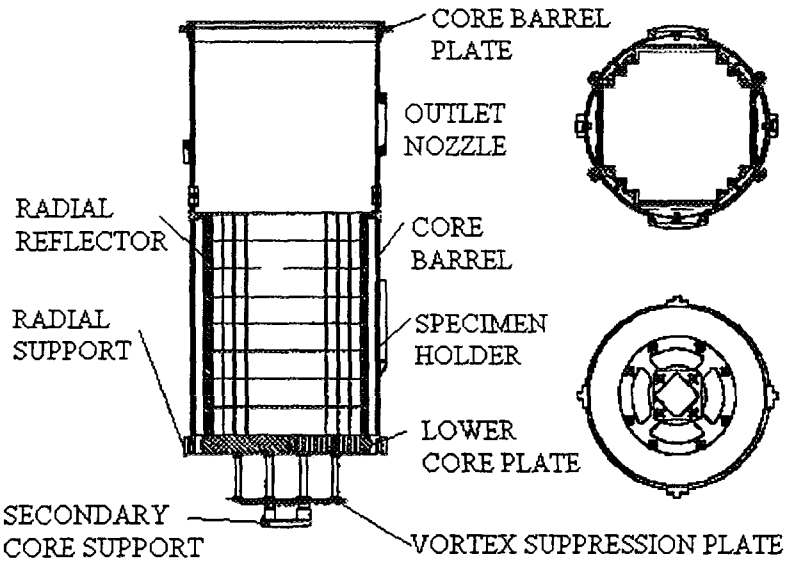


Figure 5.2-4 Lower core support assembly

In addition, the reactor internals help to direct the main coolant flow to and from the fuel assemblies; absorb control rod dynamic loads and fuel assembly loads, and transmit these loads to the reactor vessel; support instrumentation within the reactor vessel; provide protection for the reactor vessel against excessive radiation exposure from the core; and position and support reactor vessel radiation surveillance specimens.

The major containment and support member of the reactor internals is the lower core support assembly. This assembly (Figure 5.2-4) consists of the core barrel, lower core support plate, secondary core support, vortex suppression plate, radial reflectors, radial supports, and related attachment hardware. The major material for this structure is 300 series austenitic stainless steel.

The neutron reflector assemblies are located inside the core barrel and above the lower core support. These assemblies form the radial periphery of the core. Through the dimensional control of the cavity (the gap between the fuel assemblies and the reflectors) and the reflector flange holes, the reflector assemblies provide directional and metered control of the reactor coolant through the core. The reflector provides a transition from the round core barrel to the square fuel assemblies. The reflectors have longitudinal holes that allow cooling water to flow through, while still providing sufficient material to perform the neutron reflection and radiation shielding functions.

The upper core support assembly (Figure 5.2-5) consists of the upper support, the upper core plate, the support columns, and the guide tube assemblies.

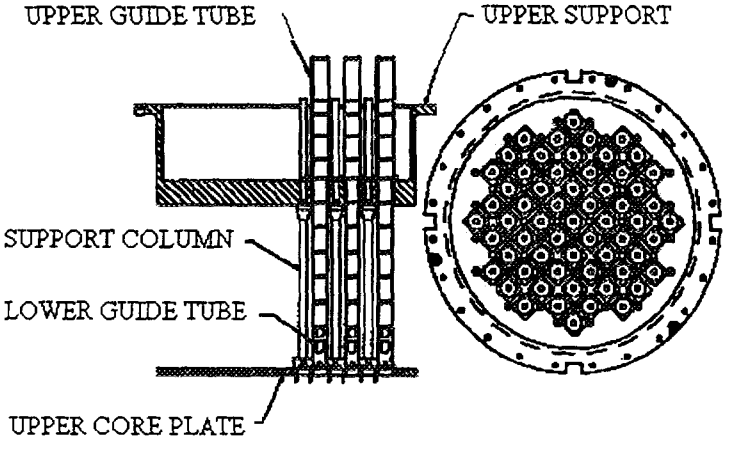


Figure 5.2-5 Upper core support assembly

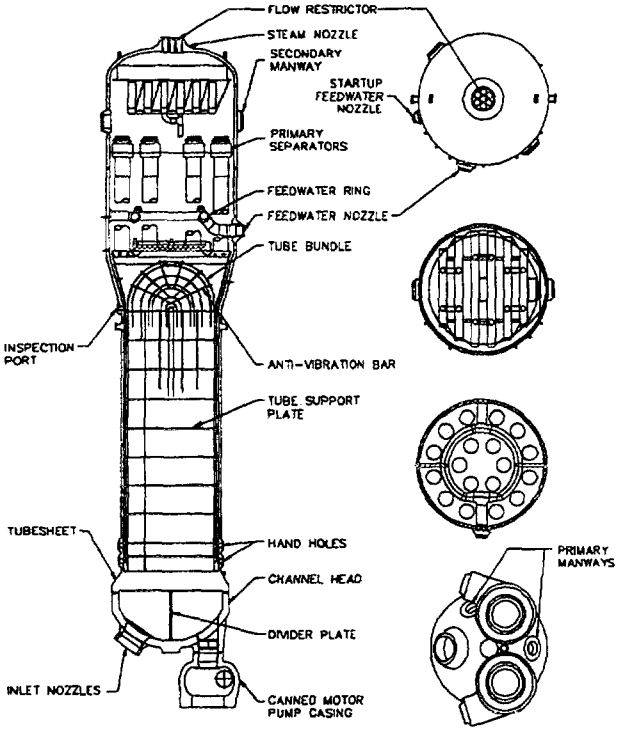


Figure 5.2-6 Steam generator

The support columns establish the spacing between the upper support and the upper core plate. The support columns are fastened at the top and bottom to these plates. The support columns transmit the mechanical loadings between the two plates and some serve the supplementary function of supporting the tubes that house the fixed in-core detectors.

The radial reflector is between the lower core barrel and core, surrounding the core and forming the core cavity. The reflector is manufactured of solid rings of stainless steel with holes bored vertically for water cooling. The stainless steel reflects fast neutrons back to the core regions. This results in lower neutron loss from the core and decreased fluence on the reactor pressure vessel. Each reflector ring is sized in height so that adjoining sections meet at a fuel grid elevation.

Steam generators

The model Delta-75 steam generator (Figure 5.2-6) is a vertical shell and U-tube evaporator with integral moisture separating equipment. The basic steam generator design and features have been proven in tests and in previous steam generators including replacement steam generator designs.

Design enhancements include nickel-chromium-iron Alloy 690 thermally treated tubes on a triangular pitch, broached tube support plates, improved antivibration bars, single-tier separators, enhanced maintenance features, and a primary-side channel head design that allows for easy access and maintenance by robotic tooling. All tubes in the steam generator are accessible for sleeving, if necessary. The design enhancements are based on proven technology.

The basic function of the steam generator is to transfer heat from the single-phase reactor coolant water through the U-shaped heat exchanger tubes to the boiling, two-phase steam mixture in the secondary side of the steam generator. The steam generator separates dry, saturated steam from the boiling mixture, and delivers the steam to a nozzle from which it is delivered to the turbine. Water from the feedwater system replenishes the steam generator water inventory by entering the steam generator through a feedwater inlet nozzle and feedring.

In addition to its steady-state performance function, the steam generator secondary side provides a water inventory which is continuously available as a heat sink to absorb primary side high temperature transients.

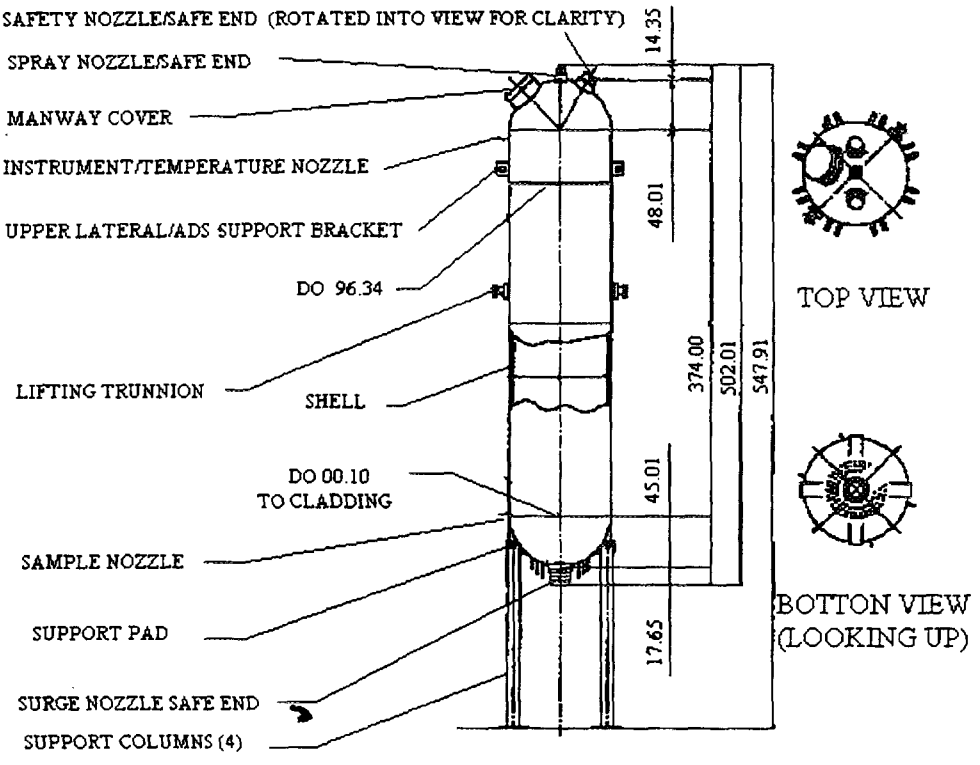


Figure 5.2-7 Pressurizer

The Delta-75 steam generator is based on standard Westinghouse Model F technology. There are currently 84 Model F steam generators operating in 25 nuclear plants with a wide range of operating environments. To date, they have accumulated over 450 steam-generator-years of operation with less than one tube plugged per year of operation. The 25 Model F-type replacement steam generators have an even more impressive record with less than one tube plugged per steam generator for every four years of operation. This is the highest level of reliability achieved by any steam generator worldwide. All the Model F-type steam generators have operated on all volatile treatment secondary side water chemistry

Pressurizer

The pressurizer (Figure 5.2-7) is of conventional design, based on proven technology and years of operating experience. The pressurizer is about 30 percent larger than that normally used in a plant of comparable power rating. The larger pressurizer increases transient operation margins, resulting in a more reliable plant with fewer reactor trips, and avoiding challenges to the plant and operator during transients. It also eliminates the need for fast-acting power-operated relief valves, which are a possible source of RCS leakage and maintenance.

The pressurizer is a principal component of the reactor coolant system pressure control system. It is a vertical, cylindrical vessel with hemispherical top and bottom heads, where liquid and vapor are maintained in equilibrium saturated conditions. A 4-inch (100 mm) spray nozzle and two 14-inch (356 mm) nozzles for connecting the safety and depressurization valve inlet headers are located in the top head. Electrical heaters are installed through the bottom head. The heaters are removable for replacement.

The bottom head contains the nozzle for attaching the surge line. This line connects the pressurizer to a hot leg, and provides for the flow of reactor coolant into and out of the pressurizer during reactor coolant system thermal expansions and contractions.

Reactor coolant pumps

The reactor coolant pumps (Figure 5.2-8) are high-inertia, high-reliability, low-maintenance, hermetically sealed canned motor pumps that circulate the reactor coolant through the reactor vessel, loop piping, and steam generators.

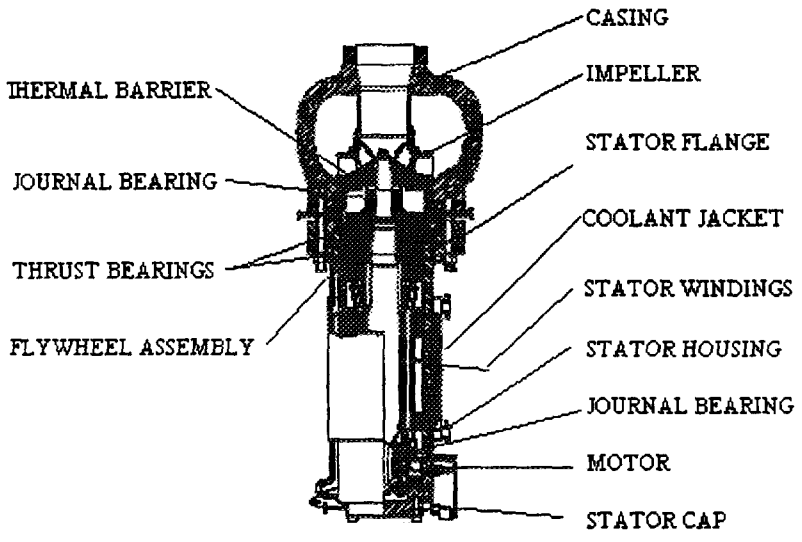


Figure 5.2-8 Reactor coolant pump

The pumps are integrated into the steam generator channel head. The integration of the pump suction into the bottom of the steam generator channel head eliminates the cross-over leg of coolant loop piping; reduces the loop pressure drop; simplifies the foundation and support system for the steam generator, pumps, and piping; and reduces the potential for uncovering of the core by eliminating the need to clear the loop seal during a small loss-of-coolant accident (LOCA). The AP-600 design uses four pumps; two pumps are coupled with each steam generator.

Since the pumps have no seals, they cannot cause a seal failure LOCA. This is a significant safety enhancement, as seal failure LOCA is a major industry issue. Maintenance is also enhanced, since seal replacement is eliminated.

The pumps are mounted in the inverted (motor-below-casing) position. Inverted canned motors have been in operation for over 28 years in marine and fossil boiler circulation systems. These pumps have better operating reliability than upright units because the motor cavity is self-venting into the pump casing, avoiding the potential for gas pockets in the bearing and water regions.

One modification of the pumps from commercial and marine canned motor pump practice is the use of a flywheel to increase the pump rotating inertia. The increased inertia provides a slower rate-of-flow coastdown to improve core thermal margins following the loss of electric power. Extensive testing has validated the manufacturability and operability of the pump flywheel assembly.

Main coolant lines

Reactor coolant system (RCS) piping is configured with two identical main coolant loops, each of which employs a single 31-inch (790 mm) inside diameter hot leg pipe to transport reactor coolant to a steam generator. The two reactor coolant pump suction nozzles are welded directly to the outlet nozzles on the bottom of the steam generator channel head. Two 22-inch (560 mm) inside diameter cold leg pipes in each loop (one per pump) transport reactor coolant back to the reactor vessel to complete the circuit.

The RCS loop layout contains several important features that provide for a significantly simplified and safer design. The reactor coolant pumps mount directly on the channel head of each steam generator. This allows the pumps and steam generator to use the same structural support, greatly simplifying the support system and providing more space for pump and steam generator maintenance. The combined steam generator/pump vertical support is a single pinned column extending from the cell floor to the bottom of the channel head. The steam generator channel head is a one-piece forging with manufacturing and inspection advantages over multipiece, welded components. The integration of the pump suction into the bottom of the steam generator channel head eliminates the crossover leg of coolant loop piping, thus avoiding the potential for core uncovery due to loop seal venting after a small loss-of-coolant accident.

The simplified, compact arrangement of the RCS also provides other benefits. The two cold leg lines of the two main coolant loops are identical (except for instrumentation and small line connections) and include bends to provide a low-resistance flow path and flexibility to accommodate the expansion difference between the hot and cold leg pipes. The piping is forged and then bent by a hot induction forming process. The use of a pipe bend reduces costs and in-service inspection requirements by eliminating welds. The loop configuration and material selection yield sufficiently low pipe stresses so that the primary loop and large auxiliary lines meet the requirements to demonstrate leak-before-break. Thus, pipe rupture restraints are not required, greatly simplifying the design and providing enhanced access for maintenance. The simplified RCS loop configuration also allows for a significant reduction in the number of snubbers, whip restraints, and supports. Field service experience and utility feedback have indicated the high desirability of these features.

5.2.2.5 Reactor auxiliary systems

Chemical and volume control system

The chemical and volume control system consists of regenerative and letdown heat exchangers, demineralizers and filters, makeup pumps, tanks, and associated valves, piping, and instrumentation.

The chemical and volume control system is designed to perform the following major tasks:

- **Purification** - maintain reactor coolant purity and activity level within acceptable limits.
- **Reactor coolant system inventory control and makeup** - maintain the required coolant inventory in the reactor coolant system; maintain the programmed pressurizer water level during normal plant operations.
- **Chemical shim and chemical control** - maintain the reactor coolant chemistry conditions by controlling the concentration of boron in the coolant for plant startups, normal dilution to compensate for fuel depletion and shutdown boration and provide the means for controlling the reactor coolant system pH by maintaining the proper level of lithium hydroxide.
- **Oxygen control** - provide the means for maintaining the proper level of dissolved hydrogen in the reactor coolant during power operation and for achieving the proper oxygen level prior to startup after each shutdown.
- **Filling and pressure testing of the reactor coolant system** - provide the means for filling and pressure testing of the reactor coolant system. The chemical and volume control system does not perform hydrostatic testing of the reactor coolant system, which is only required prior to initial startup and after major, nonroutine maintenance, but provides connections for a temporary hydrostatic test pump.
- **Borated makeup to auxiliary equipment** - provide makeup water to the primary side systems which require borated reactor grade water.
- **Pressurizer Auxiliary Spray** - provide pressurizer auxiliary spray water for depressurization.

Normal residual heat removal system

The normal residual heat removal system consists of two mechanical trains of equipment, each comprising one pump and one heat exchanger. The two trains of equipment share a common suction line from the reactor coolant system and a common discharge header. The normal residual heat removal system includes the piping, valves and instrumentation necessary for system operation. The major functions of the system are:

- **Shutdown Heat Removal.** The normal residual heat removal system removes both residual and sensible heat from the core and the reactor coolant system. It reduces the temperature of the reactor coolant system during the second phase of plant cooldown. The first phase of cooldown is accomplished by transferring heat from the reactor coolant system via the steam generators to the main steam system.

  Following cooldown, the normal residual heat removal system removes heat from the core and the reactor coolant system during the plant shutdown, until the plant is started up.

The normal residual heat removal system reduces the temperature of the reactor coolant system from 350° to 120°F (177 to 48.9 °C) within 96 hours after shutdown. The system maintains the reactor coolant temperature at or below 120°F for the plant shutdown.

- **Shutdown Purification.** The normal residual heat removal system provides reactor coolant system flow to the chemical and volume control system during refueling operations.
- **In-Containment Refuelling Water Storage Tank Cooling.** The normal residual heat removal system provides cooling for the in-containment refuelling water storage tank (IRWST) during operation of the passive residual heat removal heat exchanger or during normal plant operations when required. The system is manually initiated by the operator. The normal residual heat removal system limits the IRWST water temperature to less than 212°F (100 °C) during extended operation of the passive residual heat removal system and to not greater than 120°F during normal operation.
- **Low Pressure Reactor Coolant System Makeup and Cooling**. The normal residual heat removal system provides low pressure makeup from the IRWST to the reactor coolant system and provides additional margin for core cooling. The system is manually initiated by the operator following receipt of an automatic depressurization signal. If the system is available, it provides reactor coolant system makeup once the pressure in the reactor coolant system falls below the shutoff head of the normal residual heat removal system pumps.
- **Low Temperature Overpressure Protection.** The normal residual heat removal system provides a low temperature overpressure protection function for the reactor coolant system during refuelling, startup, and shutdown operations.
- **Long-Term, Post-Accident Containment Inventory Makeup Flowpath.** The normal residual heat removal system provides a flow path for long term post-accident makeup to the reactor containment inventory, under design assumptions of containment leakage.

5.2.2.6 Operating characteristics

The plant control scheme is based on the "reactor follows plant loads". A grid fluctuation can be compensated for through turbine control valves in case of a frequency drop. A decrease in pressure at the turbine would require an increase in reactor power.

The AP-600 is designed to withstand the following operational occurrences without the generation of a reactor trip or actuation of the safety related passive engineered safety systems:

- ± 5%/minute ramp load change within 15% and 100% power
- ± 10% step load change within 15% and 100% power
- 100% generator load rejection
- 100-50-100% power level daily load follow over 90% of the fuel cycle life
- Grid frequency changes equivalent to 10% peak-to-peak power changes at 2%/minute rate
- 20% power step increase or decrease within 10 minutes
- Loss of a single feedwater pump

The logic and setpoints for all of the AP-600 Nuclear Steam Supply System (NSSS) control systems are developed in order to meet the above operational transients without reaching any of the protection system setpoints.

5.2.3 Description of turbine generator plant system

5.2.3.1 Turbine generator plant

The AP-600 turbine consists of a double-flow, high-pressure cylinder and two double-flow, low-pressure cylinders that exhaust to individual condensers. It is a four flow tandem-compound, 1800 rpm machine. The turbine generator is intended for base load operation and also has load follow capability. Mechanical design of the turbine root and rotor steeple attachments uses optimized contour to significantly reduce operational stresses. Steam flow to the high-pressure turbine is controlled by two floor-mounted

steam chests. Each contains two throttle/stop valve assemblies, and two load-governing valves. The single moisture separator reheater (MSR) has a single stage of reheat.

The condenser and circulating water systems have been optimized. The condenser is a twin-shell, multipressure unit with one double-flow, low-pressure turbine exhausting into the top of each shell.

The turbine-generator and associated piping, valves, and controls are located completely within the turbine building. There are no safety-related systems or components located within the turbine building. The probability of destructive overspeed condition and missile generation, assuming the recommended inspection frequency, is less than 1×10^{-5} per year. In addition, orientation of the turbine-generator is such that high-energy missiles would be directed away at right angles to safety-related structures, systems, or components. Failure of turbine-generator equipment does not preclude safe shutdown of the reactor. The turbine-generator components and instrumentation associated with turbine-generator overspeed protection are accessible under operating conditions.

The single direct-driven generator is gas-cooled and rated at 856 MVA at 22 kV, and a power factor of 0.9. Other related system components include a complete turbine-generator bearing lubrication oil system, a digital electrohydraulic (DEH) control system with supervisory instrumentation, a turbine steam sealing system, overspeed protective devices, turning gear, a generator hydrogen and seal oil system, a generator CO_2 system, an exciter cooler, a rectifier section, an exciter, and a voltage regulator.

Steam from the two steam generators enters the high-pressure turbine through four stop valves and four governing control valves; two stop valves and two control valves form a single assembly. Crossties are provided upstream of the turbine stop valves to provide pressure equalization with one or more stop valves closed. After expanding through the high-pressure turbine, exhaust steam flows through one external moisture separator reheater vessel. The external moisture separator reduces the moisture content of the high-pressure exhaust steam from approximately 10-20 percent to 0.17 percent moisture or less.

The reheater uses a portion of the main steam supply to reheat the steam to superheated conditions. The reheated steam flows through separate reheat stop and intercept valves to the inlets of the low-pressure turbines. Turbine steam extraction connections are provided for seven stages of feedwater heating.

5.2.3.2 Condensate and feedwater systems

The condensate and feedwater system supplies the steam generators with heated feedwater in a closed steam cycle using regenerative feedwater heating. The condensate and feedwater system is composed of the condensate system, the main and startup feedwater system, and portions of the steam generator system

The feedwater cycle consists of seven stages of feedwater heating with two parallel string, low-pressure feedwater heaters located in the condenser neck with the next two single-string, low-pressure heaters, deaerator, and the high-pressure heaters located within the turbine building. The condenser hotwell and deaerator storage capacity allows margin in the design. This margin, coupled with three 50 percent condensate pumps, provides greater flexibility and the ability for an operator to control feedwater and condensate transients

5.2.3.3 Auxiliary systems

Radioactive waste management

The radioactive waste management systems include systems to deal with liquid, gaseous and solid waste. The liquid waste management systems include the systems that may be used to process for disposal liquids containing radioactive material. These include the following:

- Steam generator blowdown processing system
- Radioactive waste drain system
- Liquid radwaste system

The gaseous radwaste system is a once-through, ambient-temperature, charcoal delay system. The system consists of a drain pot, a gas cooler, a moisture separator, an activated charcoal-filled guard bed, and two activated charcoal-filled delay beds. Also included in the system are an oxygen analyzer subsystem and a gas sampling subsystem.

The radioactive fission gases entering the system are carried by hydrogen and nitrogen gas. The primary influent source is the liquid radwaste system degasifier. The degasifier extracts both hydrogen and fission gases from the chemical and volume control system letdown flow.

The solid waste management system is designed to collect and accumulate spent ion exchange resins and deep bed filtration media, spent filter cartridges, dry active wastes, and mixed wastes generated as a result of normal plant operation, including anticipated operational occurrences. The system is located in the auxiliary and radwaste buildings. Processing and packaging of wastes are by mobile systems in the auxiliary building loading bay and the mobile systems facility which is a part of the radwaste building. The packaged waste is stored in the annex, auxiliary and radwaste buildings until it is shipped offsite to a licensed disposal facility.

5.2.4 Instrumentation and control systems

The I&C system design for AP-600 has been assembled from existing hardware developed for nuclear plants. While the system consists of individual systems using similar technology, the heart of the system is the portions used for plant protection and for operation of the plant.

The integrated AP-600 I&C system provides the following benefits:

- Control wiring is reduced by 80 percent compared to equivalent hard wired plants without passive safety features
- Cable spreading rooms are eliminated
- Duplicate sensors, signal conditioners, and cables are eliminated
- Maintenance is simplified
- Plant design changes have little impact on I&C design
- Accurate, drift-free calibration is maintained
- Operating margins are improved

The AP-600 man-machine interfaces have been simplified compared to existing plants. The probability of operator error is reduced and operations, testing, and maintenance are simplified. An automatic signal selector in the control system selects from a redundant sensor for control inputs in lieu of requiring manual selection by the control board operator. Accident monitoring and safety parameters are displayed on safety qualified displays with a co-ordinated set of graphics generated by the qualified data processor. The major benefits of the improved man-machine interfaces are:

- Reduced quantity of manual actions is required
- Reduced quantity of data is presented to operator
- Number of alarms is reduced
- Improved quality of data is presented to operator
- Data is interpreted for the operator by system computer
- Maintenance is simplified

5.2.4.1 Design concept, including control room

Figure 5.2-9 illustrates the instrumentation and control architecture for the AP-600. It is arranged in a hierarchical manner to provide a simplified, structured design that is horizontally and vertically integrated.

Above the monitor bus are the systems that facilitate the interaction between the plant operators and the I&C. These are the operations and control centers system (OCS) and the data display and monitoring system (DDS). Below the monitor bus are the systems and functions that perform the protective, control, and data monitoring functions.

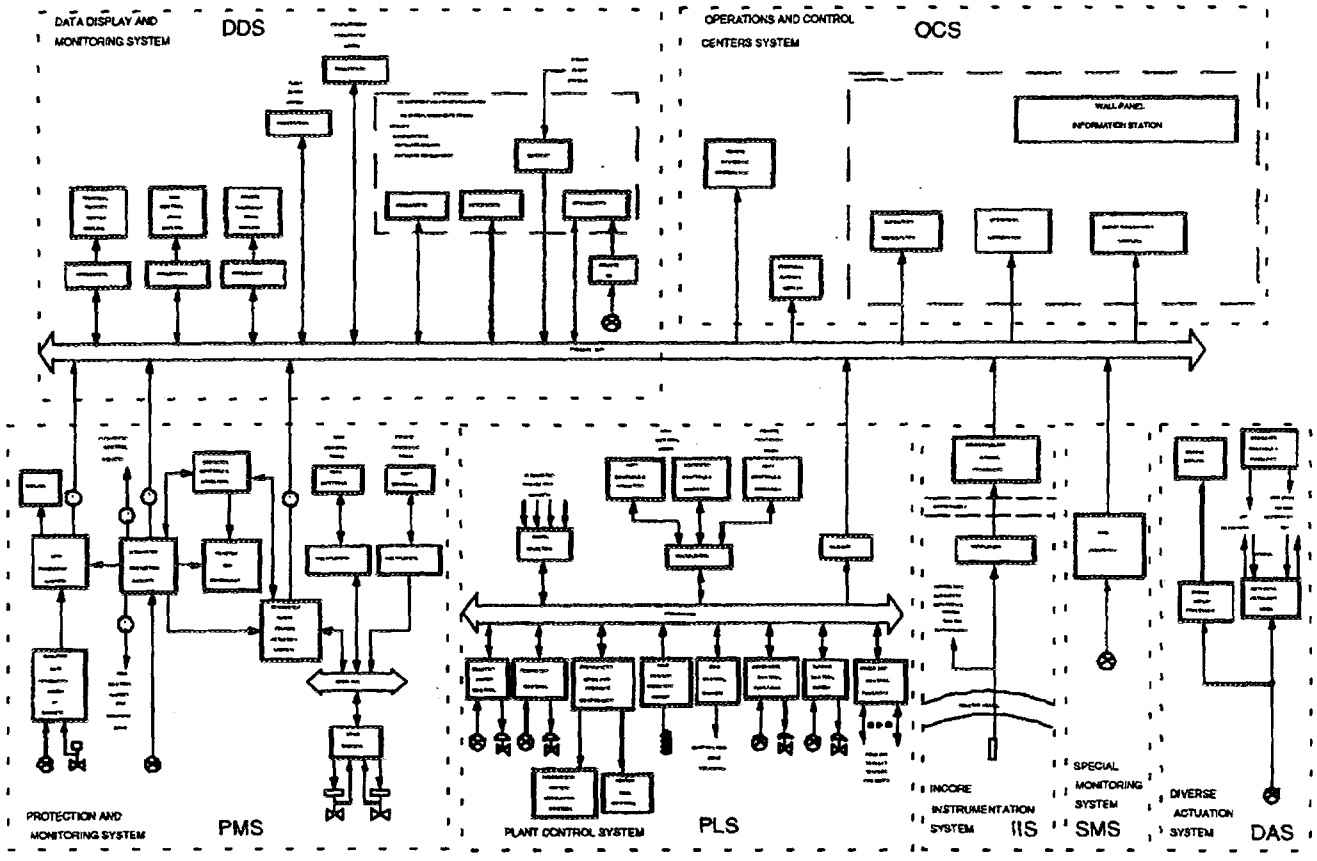


Figure 5.2-9 Instrumentation and control architecture

These are the protection and safety monitoring system (PMS) (Section 5.2.4.2) the plant control system (PLS), the special monitoring system (SMS), and the in-core instrumentation system (IIS).

The plant control system (PLS) has the function of establishing and maintaining the plant operating conditions within prescribed limits. The control system improves plant safety by minimizing the number of situations for which some protective response is initiated and relieves the operator from routine tasks.

The purpose of the diverse actuation system (DAS) is to provide alternative means of initiating the reactor trip and emergency safety features. The hardware and software used to implement the DAS are different from the hardware and software used to implement the protection and safety monitoring system. The DAS is included to meet the anticipated transient without (reactor) trip (ATWT) rule and to reduce the probability of a severe accident resulting from the unlikely coincidence of a transient and common mode failure of the protection and safety monitoring. The protection and safety monitoring system is designed to prevent common mode failures; however, in the low-probability case where a common mode failure could occur, the DAS provides diverse protection.

Main control room

The operations and control centers system includes the complete operational scope of the main control room, the remote shutdown workstation, the waste processing control room, and partial scope for the technical support center. With the exception of the control console structures, the equipment in the control room is part of the other systems (for example, protection and safety monitoring system, plant control system, data and display processing system). The conceptual arrangement of the main control room is shown in Figure 5.2-10.

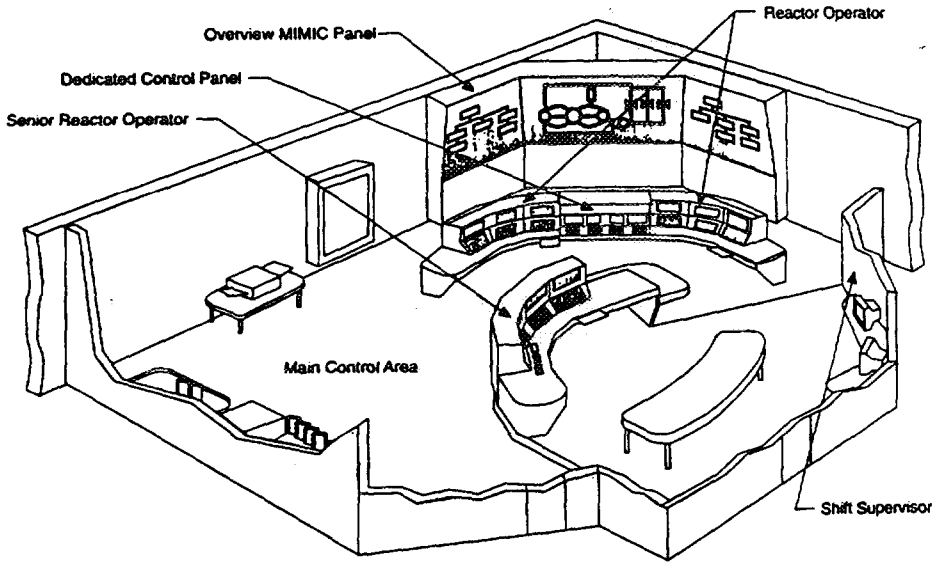


Figure 5.2-10 Main control room

The boundaries of the operations and control center system for the main control room and the remote shutdown workstation are the signal interfaces with the plant components. These interfaces are via the plant protection and safety monitoring system processor and logic circuits, which interface with the reactor trip and engineered safety features plant components; the plant control system processor and logic circuits, which interface with the nonsafety-related plant components; and the plant monitor bus, which provides plant parameters, plant component status, and alarms.

5.2.4.2 Reactor protection system and other safety systems

The AP-600 provides instrumentation and controls to sense accident situations and initiate engineered safety features. The occurrence of a limiting fault, such as a loss-of-coolant accident or a secondary system break, requires a reactor trip plus actuation of one or more of the engineered safety features. This combination of events prevents or mitigates damage to the core and reactor coolant system components, and provides containment integrity.

The protection and safety monitoring system (PMS) provides the safety-related functions necessary to control the plant during normal operation, to shut down the plant, and to maintain the plant in a safe shutdown condition. The protection and safety monitoring system controls safety-related components in the plant that are operated from the main control room or remote shutdown workstation.

5.2.5 Electrical systems

The AP-600 on-site power system includes the main AC power system and the DC power system. The main AC power is a non-Class 1E system. The DC power system consists of two independent systems, one Class 1E and one non-Class 1E. The on-site power system is designed to provide reliable electric power to the plant safety and non-safety equipment for normal plant operation, startup, and normal shut down, and for accident mitigation and emergency shutdown.

The main generator is connected to the off-site power system via three single-phase main step-up transformers. The normal power source for the plant auxiliary AC loads is provided from the 22 kV isophase generator buses through the two unit auxiliary transformers of identical ratings. In the event of a loss of the main generator, the power is maintained without interruption from the preferred power supply by an auto-trip of the main generator breaker. Power then flows from the transformer area to the auxiliary loads through the main and unit auxiliary transformers.

Off-site power has no safety-related function due to the passive safety features incorporated in the AP-600 design. Therefore, redundant off-site power supplies are not required. The design provides a reliable offsite power system that minimizes challenges to the passive safety system.

5.2.5.1 Operational power supply systems

The main AC power system is a non-Class 1E system that does not perform any safety function. The standby power supply is included in the on-site standby power system.

The power to the main AC power system normally comes from the station main generator through unit auxiliary transformers. The plant is designed to sustain a load rejection from 100 percent power with the turbine generator continuing stable operation while supplying the plant house loads. The load rejection feature does not perform any safety function

The on-site standby AC power system is powered by the two on-site standby diesel generators and supplies power to selected loads in the event of loss of normal, and preferred AC power supplies.

The plant DC power system comprises two independent Class 1E and non-Class 1E DC power systems. Each system consists of ungrounded stationary batteries, DC distribution equipment, and uninterruptible power supplies.

5.2.5.2 Safety-related systems

The Class 1E DC power system includes four independent divisions of battery systems. Any three of the four divisions can shut down the plant safely and maintain it in a safe shutdown condition. Divisions B and C have two battery banks. One of these battery banks is sized to supply power to selected safety-related loads for at least 24 hours, and the other battery bank is sized to supply power to another set of selected safety-related loads for at least 72 hours following a design basis event (including the loss of all AC power).

5.2.6 Safety concepts

5.2.6.1 Safety requirements and design philosophy

The AP-600 design provides for multiple levels of defense for accident mitigation (defense-in-depth), resulting in extremely low core damage probabilities while minimizing the occurrences of containment flooding, pressurization, and heat-up situations. This defense-in-depth capability includes multiple levels of defense for a very wide range of plant events. Defense-in-depth is integral to the AP-600 design, with a multitude of individual plant features capable of providing some degree of defense of plant safety. Six aspects of the AP-600 design contribute to defense-in-depth:

Stable Operation. In normal operation, the most fundamental level of defense-in-depth ensures that the plant can be operated stably and reliably. This is achieved by the selection of materials, by quality assurance during design and construction, by well-trained operators, and by an advanced control system and plant design that provide substantial margins for plant operation before approaching safety limits.

Physical Plant Boundaries. One of the most recognizable aspects of defense-in-depth is the protection of public safety through the physical plant boundaries. Releases of radiation are directly prevented by the fuel cladding, the reactor pressure boundary, and the containment pressure boundary. For the fuel cladding boundary, the reactor protection system is designed to actuate a reactor trip whenever necessary to prevent exceeding the fuel design limits. The core design, together with defense-in-depth process and decay heat removal systems, provides this capability under expected conditions of normal operation, with appropriate margin for uncertainties and anticipated transient situations. The reactor coolant pressure boundary is designed with complete overpressure protection and appropriate materials to provide and maintain the boundary during all modes of plant operation. The containment vessel, in conjunction with the defense-in-depth heat removal systems, is designed so that its design pressure is not exceeded following

postulated design basis accidents, and containment failure does not occur even under severe accident conditions.

Passive Safety-Related Systems. The highest level of defense includes the AP-600 safety-related passive systems and equipment. The safety-related passive systems are sufficient to automatically establish and maintain core cooling and containment integrity for the plant following design basis events, assuming that the most limiting single failure occurs. These systems maintain core cooling and containment integrity after an event, without operator action and onsite and offsite ac power sources, for an indefinite amount of time.

The safety-related passive systems use only natural forces, such as gravity, natural circulation, and compressed gas for system operation. No pumps, fans, diesels, chillers, or other rotating machinery are used. A few simple valves align the passive safety systems when they are automatically actuated by the safety-related protection and safety monitoring system (PMS). The PMS provides the safety-related functions of reactor trip, engineered safeguards features actuation, and post-accident monitoring.

Diversity within the Safety-Related Systems. An additional level of defense is provided through the diverse mitigation functions within the passive safety-related systems themselves. This diversity exists, for example, in the residual heat removal function. The PRHR HX is the passive safety-related feature for removing decay heat during a transient. In case of multiple failures in the PRHR HX, defense-in-depth is provided by the passive safety injection and automatic depressurization (passive feed and bleed) functions of the PXS.

Non-safety Systems. The next level of defense-in-depth is the availability of certain non-safety systems for reducing the potential for events leading to core damage. For more probable events, these defense-in-depth non-safety systems automatically actuate to provide a first level of defense to reduce the likelihood of unnecessary actuation and operation of the safety-related systems. These non-safety-related systems establish and maintain safe shutdown conditions for the plant following design basis events, provided that at least one of the nonsafety-related ac power sources is available.

Containing Core Damage. The AP-600 design provides the operators with the ability to drain the in-containment refuelling water storage tank (IRWST) water into the reactor cavity in the event that the core has uncovered and is melting. The objective of this cavity flooding action is to prevent reactor vessel failure and subsequent relocation of molten core debris into the containment. Retention of the debris in the vessel significantly reduces the uncertainty in the assessment of containment failure and radioactive release to the environment due to ex-vessel severe accident phenomena.

AP-600 defense-in-depth features enhance safety such that no severe release of fission products is predicted to occur from an initially intact containment for more than 100 hours after the onset of core damage, assuming no actions for recovery. This amount of time provides for performance of accident management actions to mitigate the accident and prevent containment failure. The frequency of severe release as predicted by PRA is 3.0×10^{-8} per reactor year, which is much lower than for conventional plants.

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

The use of passive safety systems provides superiority over conventional plant designs through significant and measurable improvements in plant simplification, safety, reliability, and investment protection. The AP-600 uses passive safety systems to improve the safety of the plant and to satisfy safety criteria of regulatory authorities. The passive safety systems require no operator actions to mitigate design basis accidents. These systems use only natural forces such as gravity, natural circulation, and compressed gas to make the systems work. No pumps, fans, diesels, chillers, or other active machinery are used. A few simple valves align and automatically actuate the passive safety systems. To provide high reliability, these valves are designed to actuate to their safeguards positions upon loss of power or upon receipt of a safeguards actuation signal. However, they are also supported by multiple, reliable power sources to avoid unnecessary actuations.

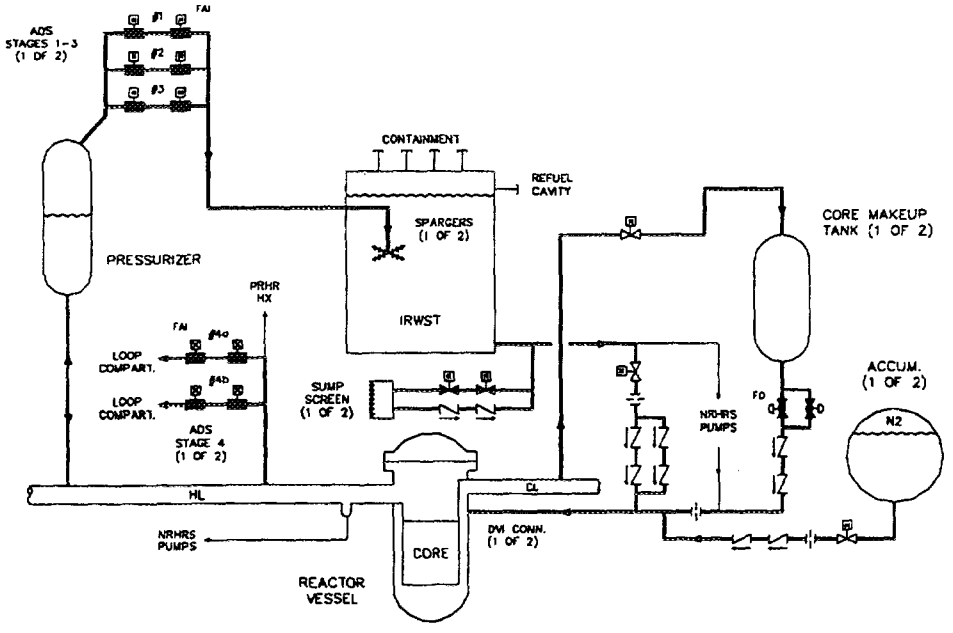


Figure 5.2-11 Passive core cooling system

The AP-600 passive safety-related systems include:

- The passive core cooling system (PXS)
- The passive containment cooling system (PCS)
- The main control room habitability system (VES)
- Containment isolation

These passive safety systems provide a major enhancement in plant safety and investment protection as compared with conventional plants. They establish and maintain core cooling and containment integrity indefinitely, with no operator or ac power support requirements. The passive systems are designed to meet the single-failure criteria, and probabilistic risk assessments (PRAs) are used to verify their reliability.

The AP-600 passive safety systems are significantly simpler than typical PWR safety systems since they contain significantly fewer components, reducing the required tests, inspections, and maintenance, require no active support systems, and their readiness is easily monitored. The passive safety systems have one-third the number of remote valves as typical active safety systems, and they contain no pumps. In addition, these passive safety systems do not require a radical departure in the design of the rest of the plant, core, RCS, or containment.

The passive safety systems do not require the large network of active safety support systems that are needed in typical nuclear plants. These systems include ac power, HVAC, cooling water, and the associated seismic buildings to house these components. This simplification applies to the emergency diesel generators and their network of support systems, air start, fuel storage tanks and transfer pumps, and the air intake/exhaust system. As a result, these support systems no longer must be safety class, and they are simplified or eliminated. For example, the essential service water system and its associated safety cooling towers are replaced with a non-safety service water cooling system. In the AP-600 plant, the non-safety support systems and the passive safety systems have been integrated into the plant design by systematically optimizing the plant layout for these systems. This design approach allows the licensing safety criteria to be satisfied with a greatly simplified plant design.

Emergency core cooling system

The passive core cooling system (PXS) (Figure 5.2-11) protects the plant against reactor coolant system (RCS) leaks and ruptures of various sizes and locations. The PXS provides the safety functions of core residual heat removal, safety injection, and depressurization. Safety analyses (using US NRC-approved codes) demonstrate the effectiveness of the PXS in protecting the core following various RCS break events,

even for breaks as severe as the 8-inch (200 mm) vessel injection lines. The PXS provides approximately a 400°F (220 °C) margin to the maximum peak clad temperature limit for the double-ended rupture of a main reactor coolant pipe.

Safety injection and depressurization

The PXS uses three passive sources of water to maintain core cooling through safety injection. These injection sources include the core makeup tanks (CMTs), the accumulators, and the in-containment refuelling water storage tank (IRWST). These injection sources are directly connected to two nozzles on the reactor vessel so that no injection flow can be spilled for the larger break cases.

Long-term injection water is provided by gravity from the IRWST, which is located in the containment just above the RCS loops. Normally, the IRWST is isolated from the RCS by check valves. This tank is designed for atmospheric pressure. As a result, the RCS must be depressurized before injection can occur.

The depressurization of the RCS is automatically controlled to reduce pressure to about 12 psig (0.18 MPa); at which point, the head of water in the IRWST overcomes the low RCS pressure and the pressure loss in the injection lines. The PXS provides for depressurization using the four stages of the ADS to permit a relatively slow, controlled RCS pressure reduction.

Passive residual heat removal

The PXS includes a 100% capacity passive residual heat removal heat exchanger (PRHR HX) (Figure 5.2-12). The PRHR HX is connected through inlet and outlet lines to RCS loop 1. The PRHR HX protects the plant against transients that upset the normal steam generator feedwater and steam systems. The PRHR HX satisfies the safety criteria for loss of feedwater, feedwater line breaks, and steam line breaks.

The IRWST provides the heat sink for the PRHR HX. The IRWST water volume is sufficient to absorb decay heat for more than 1 hour before the water begins to boil. Once boiling starts, steam passes to the containment. This steam condenses on the steel containment vessel and, after collection, drains by gravity back into the IRWST. The PRHR HX and the passive containment cooling system provide indefinite decay heat removal capability with no operator action required.

Passive containment cooling system

The passive containment cooling system (PCS) (Figure 5.2-13) provides the safety-related ultimate heat sink for the plant. As demonstrated by computer analyses and extensive test programs, the PCS effectively cools the containment following an accident such that the design pressure is not exceeded and the pressure is rapidly reduced.

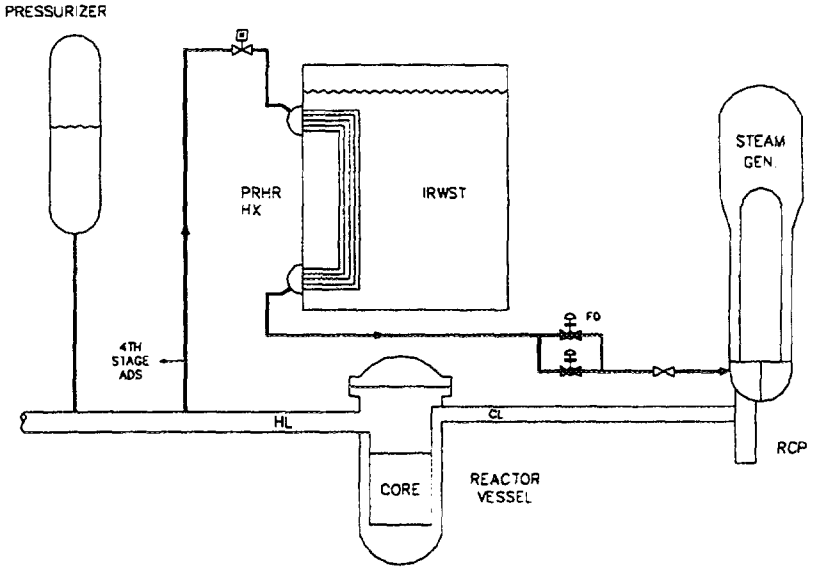


Figure 5.2-12 Passive residual heat removal system

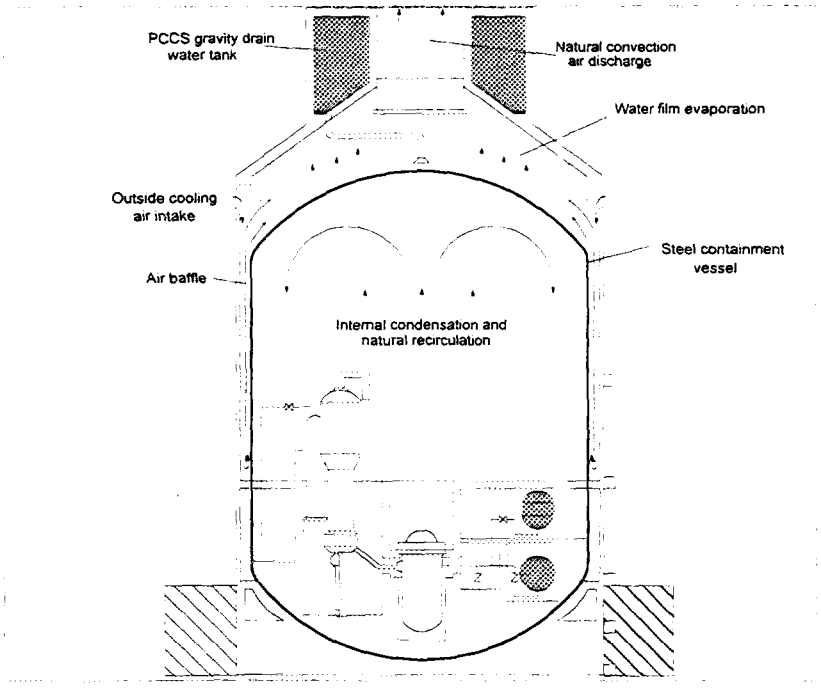


Figure 5.2-13 Passive containment cooling system

The steel containment vessel provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Heat is removed from the containment vessel by continuous natural circulation flow of air. During an accident, the air cooling is supplemented by evaporation of water. The water drains by gravity from a tank located on top of the containment shield building.

Westinghouse has calculated the AP-600 to have a significantly reduced frequency of release of large amounts of radioactivity following a severe accident core damage scenario. This analysis shows that with only the normal PCS air cooling, the containment stays well below the predicted failure pressure. Other factors include improved containment isolation and reduced potential for LOCAs outside of containment. This improved containment performance supports the technical basis for simplification of offsite emergency planning.

Main control room habitability system

The main control room habitability system (VES) provides fresh air, cooling, and pressurization to the main control room (MCR) following a plant accident. Operation of the VES is automatically initiated upon receipt of a high MCR radiation signal, which isolates the normal control room ventilation path and initiates pressurization. Following system actuation, all functions are completely passive.

The VES air supply is contained in a set of compressed air storage tanks. The VES also maintains the MCR at a slight positive pressure, to minimize the infiltration of airborne contaminants from the surrounding areas.

Containment isolation

AP-600 containment isolation is significantly improved over that of conventional PWRs. One major improvement is the large reduction in the number of penetrations. Furthermore, the number of normally open penetrations is reduced by 60 percent. There are no penetrations required to support post-accident mitigation functions (the canned motor reactor coolant pumps do not require seal injection, and the passive residual heat removal and passive safety injection features are located entirely inside containment).

Long-term accident mitigation

A major safety advantage of the AP600 versus current-day PWRs is that long-term accident mitigation is maintained without operator action and without reliance on offsite or onsite ac power sources. Existing plants rely on operator actions for both short-term and long-term mitigation and are powered from

either onsite or offsite ac power sources. The passive safety systems are designed to provide long-term core cooling and decay heat removal without the need for operator actions and without reliance on the active nonsafety-related systems.

For the limiting design basis accidents, the core coolant inventory in the containment for recirculation cooling and boration of the core is sufficient to last for at least 30 days, even if inventory is lost at the design basis containment leak rate.

5.2.6.3 Severe accidents (Beyond design basis accidents)

In-vessel retention of molten core debris

In-vessel retention (IVR) of core debris by cooling from the outside is a severe accident mitigation attribute of the AP-600 design. With the reactor vessel intact and debris retained in the lower head, phenomena that may occur as a result of core debris being relocated to the reactor cavity are prevented. The AP-600 is provided with reactor vessel insulation that promotes in-vessel retention and surface treatment that promotes wetability of the external surface.

The design features of the AP-600 containment promote flooding of the containment cavity region during accidents, submerging of the reactor vessel lower head in water. Liquid effluent released through the break during a LOCA event is directed to the reactor cavity. The AP-600 design also includes a provision for draining the in-containment refuelling water storage tank (IRWST) water into the reactor cavity through an operator action.

5.2.7 Plant layout

5.2.7.1 Buildings and structures, including plot plan

A typical site plan for a single unit AP-600 is shown on Figure 5.2-14. The power block complex consists of five principal building structures; the nuclear island, the turbine building, the annex building, the diesel generator building and the radwaste building. Each of these building structures are constructed on individual basemats. The nuclear island consists of the containment building, the shield building, and the auxiliary building, all of which are constructed on a common basemat.

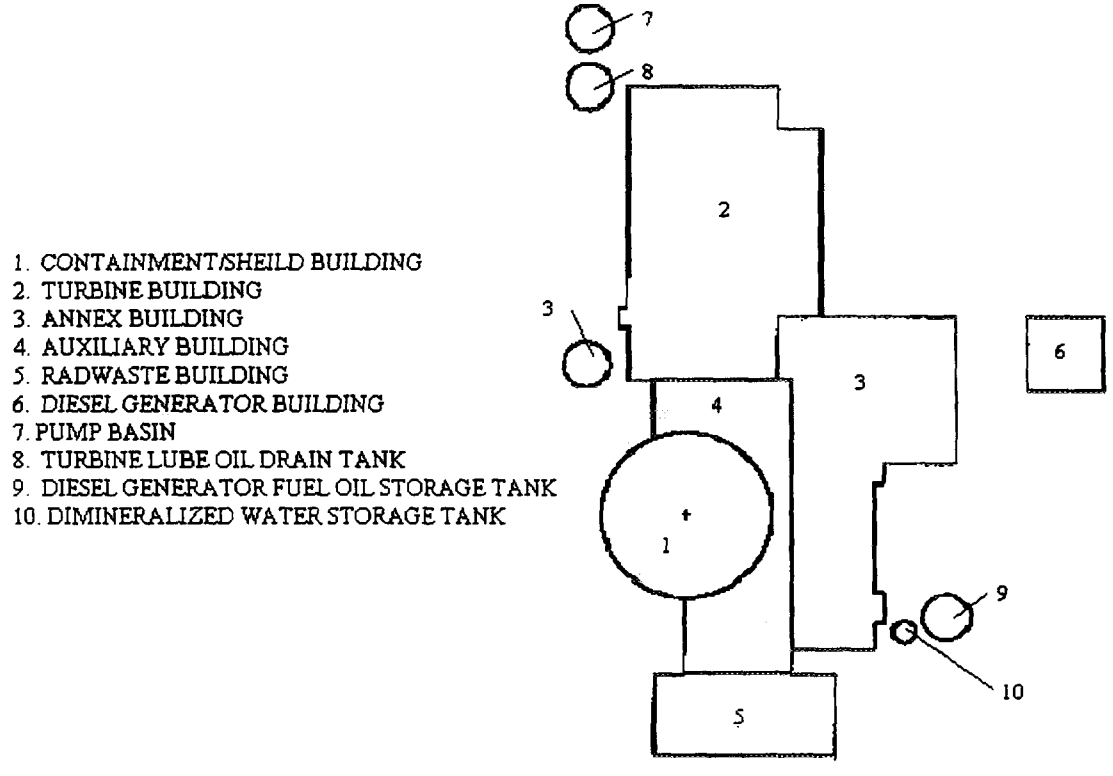


Figure 5.2-14 AP-600 - Site layout

Plant arrangement

The plant arrangement contains conventional and unique features that facilitate and simplify operational and maintenance activities. For example, accessibility to the containment during an outage is extremely important to those maintenance activities that can be performed only during the outage. A conventional containment may have only one large equipment hatch, a personnel airlock, and an emergency escape hatch. The AP-600 containment contains a 22-foot (6.7 m) diameter main equipment hatch and a personnel airlock at the operating deck level, and a 16-foot (4.9 m) diameter maintenance hatch and a personnel airlock at grade level. These large hatches significantly enhance accessibility to the containment during outages and, consequently, reduce the potential for congestion at the containment entrances. These containment hatches, located at the two different levels, allow activities occurring above the operating deck to be unaffected by activities occurring below the operating deck.

The containment arrangement provides significantly larger laydown areas inside containment than most conventional plants at both the operating deck level and the maintenance floor level. Additionally, the auxiliary building and the adjacent annex building provide large staging and laydown areas immediately outside of both large equipment hatches.

The AP-600 consists of the following five principal structures. Each of these buildings are constructed on individual basemats:

- Nuclear island (Containment, Shield building, Auxiliary building)
- Annex building
- Diesel generator building
- Radwaste building
- Turbine building

5.2.7.2 Reactor building

The reactor building of the AP-600 basically coincides with the shield building surrounding the containment (Section 5.2.7.3).

5.2.7.3 Containment

Containment building

The containment building is the containment vessel and all structures contained within the containment vessel. The containment building is an integral part of the overall containment system with the functions of containing the release of airborne radioactivity following postulated design basis accidents and providing shielding for the reactor core and the reactor coolant system during normal operations.

The containment vessel is an integral part of the passive containment cooling system. The containment vessel and the passive containment cooling system are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents.

The principal system located within the containment building is the reactor coolant system that consists of two main coolant loops, a reactor vessel, two steam generators, four canned motor reactor coolant pumps, and a pressurizer.

The passive core cooling system is also located in the containment building. The primary components of the passive core cooling system are two core makeup tanks, two accumulators, the in-containment refuelling water storage tank, the passive residual heat removal heat exchanger, and two spargers. The automatic depressurization valves are located above the pressurizer and consist of a two-tier valve module.

Shield building

The shield building is the structure and annulus area that surrounds the containment building. During normal operations, a primary function of the shield building is to provide shielding for the containment

and the radioactive systems and components located in the containment building. The shield building, in conjunction with the internal structures of the containment building, provides the required shielding for the reactor coolant system and all the other radioactive systems and components housed in the containment. During accident conditions, the shield building provides the required shielding for radioactive airborne materials that may be dispersed in the containment as well as radioactive particles in the water distributed throughout the containment.

The shield building is also an integral part of the passive containment cooling system. The passive containment cooling system air baffle is located in the upper annulus area. It is attached to the cylindrical section of the containment vessel. The function of the passive containment cooling system air baffle is to provide a pathway for natural circulation of cooling air in the event that a design basis accident results in a large release of energy into the containment. In this event the outer surface of the containment vessel transfers heat to the air between the baffle and the containment shell. This heated and thus, lower density air flows up through the air baffle to the air diffuser and cooler and higher density air is drawn into the shield building through the air inlet in the roof of the shield building.

Another function of the shield building is to protect the containment building from external events. The shield building protects the containment vessel and the reactor coolant system from the effects of tornadoes and tornado produced missiles.

5.2.7.4 Turbine building

The turbine building houses the main turbine, generator, and associated fluid and electrical systems. It provides weather protection for the laydown and maintenance of major turbine/generator components. The turbine building also houses the makeup water purification system. No safety-related equipment is located in the turbine building.

5.2.7.5 Other buildings

Auxiliary building

The primary function of the auxiliary building is to provide protection and separation for the seismic Category I mechanical and electrical equipment located outside the containment building. The auxiliary building provides protection for the safety-related equipment against the consequences of either a postulated internal or external event. The auxiliary building also provides shielding for the radioactive equipment and piping that is housed within the building.

The most significant equipment, systems and functions contained within the auxiliary building are the following:

Main control room. The main control room provides the man-machine interfaces required to operate the plant safely under normal conditions and to maintain it in a safe condition under accident conditions.

Instrumentation and control systems. The instrumentation and control systems provide monitoring and control of the plant during startup, ascent to power, powered operation, and shutdown.

Class 1E direct current system. The Class 1E DC system provides 125 volts power for safety-related and vital control instrumentation loads including monitoring and control room emergency lighting

Fuel handling area. The primary function of the fuel handling area is to provide for the handling and storage of new and spent fuel

Mechanical equipment areas. The mechanical equipment located in radiological control areas of the auxiliary building are the normal residual heat removal pumps and heat exchangers, the spent fuel cooling system pumps and heat exchangers, the liquid and gaseous radwaste pumps, tanks, demineralizers and filters, the chemical and volume control pumps and heating, ventilating and air conditioning exhaust fans.

Containment penetration areas. The auxiliary building contains all of the containment penetration areas from the shield building for mechanical, electrical and instrumentation and control penetrations

Main steam and feedwater isolation valve compartment. The main steam and feedwater isolation valve compartment is contained within the auxiliary building. The auxiliary building provides an adequate venting area for the main steam and feedwater isolation valve compartment in the event of a postulated leak in either a main steam line or feedwater line.

Annex building

The annex building provides the main personnel entrance to the power generation complex. It includes access ways for personnel and equipment to the clean areas of the nuclear island in the auxiliary building and to the radiological control area. The building includes the health physics facilities for the control of entry to and exit from the radiological control area as well as personnel support facilities such as locker rooms.

Diesel generator building

The diesel generator building houses two identical slide along diesel generators separated by a three hour fire wall. These generators provide backup power for plant operation in the event of disruption of normal power sources. No safety-related equipment is located in the diesel generator building.

Radwaste building

The radwaste building includes facilities for segregated storage of various categories of waste prior to processing, for processing by mobile systems, and for storing processed waste in shipping and disposal containers. No safety-related equipment is located in the radwaste building.

5.2.8 Technical data

General plant data		
Power plant output, gross	619	MWe
Power plant output, net	600	MWe
Reactor thermal output (core power 1933 MWt)	1 940	MWt
Power plant efficiency, net	35	%
Cooling water temperature	30.5	°C
Nuclear steam supply system		
Number of coolant loops	2 hot legs/4 cold legs	
Primary circuit volume, including pressuriser	239	m^3
Steam flow rate at nominal conditions	1 063	kg/s
Feedwater flow rate at nominal conditions	1 063	kg/s
Steam temperature/pressure	272.7/5.74	°C/MPa
Feedwater temperature/pressure	285.0/7.21	°C/MPa
Reactor coolant system		
Primary coolant flow rate	9 190	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	279.5	°C
Coolant outlet temperature, at RPV outlet	315.6	°C
Mean temperature rise across core	36.1	°C
Reactor core		
Active core height	3.658	m
Equivalent core diameter	2.921	m
Heat transfer surface in the core	4 170	m^2
Fuel inventory	66.9	t U
Average linear heat rate	13.5	kW/m
Average fuel power density	28.89	kW/kg U
Average core power density (volumetric)	78.82	kW/l
Thermal heat flux, F_q	2.60	kW/m^2
Enthalpy rise, F_H	1.65	

Fuel material	Sintered UO_2	
Fuel assembly total length	4 326	mm
Rod array	square, 17×17	
Number of fuel assemblies	145	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	25	
Number of structural spacer grids	9	
Number of intermediate flow mixing grids	4	
Enrichment (range) of first core	1.9-3.7	Wt% U-235
Enrichment of reload fuel at equilibrium core	4.8	Wt% U-235
Operating cycle length (fuel cycle length)	24	months
Average discharge burnup of fuel (nominal)	55 000	MWd/t
Cladding tube material	Zircaloy	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly	664.5	kg
Active length of fuel rods	4 094	mm
Burnable absorber, strategy/material	Wet annular burnable absorber, Integral fuel burnable absorber	
Number of control rods	61 (45 black, 16 gray)	
Absorber rods per control assembly	24	
Absorber material	Ag-In-Cd (black), Ag-In-Cd/304SS (gray)	
Drive mechanism	Magnetic jack	
Positioning rate [in steps/min or mm/s]	45	steps/min
Soluble neutron absorber	Boric acid	
Reactor pressure vessel		
Cylindrical shell inner diameter	3 988	mm
Wall thickness of cylindrical shell	203	mm
Total height	11 708	mm
Base material: cylindrical shell	Carbon steel	
RPV head	Carbon steel	
liner	Stainless steel	
Design pressure/temperature	17.1/ 360	MPa/°C

Transport weight (lower part), and	283.3	t
RPV head	79.5	t

Steam generators

Type	Delta 75, vertical, U-tube	
Number [Thermal capacity 970 MWt]	2	
Heat transfer surface	6 986	m^2
Number of heat exchanger tubes	6 307	
Tube dimensions	17.5/15.5	mm
Maximum outer diameter	4500.8	mm
Total height	21051	mm
Transport weight	365.5	t
Shell and tube sheet material	Carbon steel	
Tube material	Inconel 690-TT	

Reactor coolant pump

Type	Canned motor	
Number	4	
Design pressure/temperature	17.1/343.3	MPa/°C
Design flow rate (at operating conditions)	4 970	kg/s
Pump head	73	m
Power demand at coupling, cold/hot	2 240	kW
Pump casing material		
Pump speed		rpm

Pressuriser

Total volume	45.31	m^3
Steam volume: full power/zero power	14.16	m^3
Design pressure/temperature	17.1/360	MPa/°C
Heating power of the heater rods	1600	kW
Number of heater rods		
Inner diameter	354	mm
Total height		mm
Material		
Transport weight		t

Pressuriser relief tank

Not applicable

Total volume	m^3
Design pressure/temperature	MPa/°C
Inner diameter (vessel)	mm
Total height	mm
Material	
Transport weight	t

Primary containment

Type	Dry, free standing, steel cylindrical	
Overall form (spherical/cyl.)		
Dimensions (diameter/height)	39.6/57.6	m
Free volume		m^3
Design pressure/temperature (DBEs)	0.316/137.8	kPa/°C
(severe accident situations)	0.316 /137.8	kPa/°C
Design leakage rate	0.12	vol%/day
Is secondary containment provided?	No	

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage	22/	kV
	rated capacity	870	MVA
Plant transformers,	rated voltage	22/4.16	kV
	rated capacity	45	MVA
Start-up transformer	rated voltage	-/4.16	kV
	rated capacity	45	MVA
Medium voltage busbars (6 kV or 10 kV)		6	
Number of low voltage busbar systems		10	
Standby diesel generating units: number		2	
	rated power	4	MW
Number of diesel-backed busbar systems		2	
Voltage level of these		4160	V ac
Number of DC distributions		10	

Voltage level of these	125	V dc
Number of battery-backed busbar systems	11	
Voltage level of these	125	V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	Tandem-compound, 4-flow, 47 in. (1200 mm) last-stage blade	
Number of turbine sections per unit	1HP/ 2LP	
Turbine speed	1 800	rpm
Overall length of turbine unit	30	m
Overall width of turbine unit	9	m
HP inlet pressure/temperature	5.6/271.4	MPa/°C

Generator

Type	3-phase, synchronous	
Rated power	880	MVA
Active power	675	MW
Voltage	22	kV
Frequency	60	Hz
Total generator mass [1,216,000 lbs]	552	t
Overall length of generator	18	m

Condenser

Type	Multipressure	
Number of tubes	50 600	
Heat transfer area	73 784	m^2
Cooling water flow rate	24.36	m^3/s
Cooling water temperature	30.5	°C
Condenser pressure	9.1	kPa

Condensate pumps

Number	3	
Flow rate	389	kg/s
Pump head	267	m
Temperature	46	°C
Pump speed	1190	rpm

Condensate clean-up system

Full flow/part flow	part flow, 33%
Filter type	Deep bed

Feedwater tank

Volume	284	m^3
Pressure/temperature	1.11/184	MPa/°C

Feedwater pumps

Number	2	
Flow rate	590	kg/s
Pump head	783	
Feedwater temperature	184	°C
Pump speed	4300	rpm

Condensate and feedwater heaters

Number of heating stages	7
Redundancies	Two strings for lowest two stages

5.2.9 Project status and planned schedule

The AP-600 is scheduled to obtain final design approval from the NRC in 1997 and design certification in 1998. The standard safety analysis report (SSAR) was submitted to NRC for review on June 26, 1992. Systems and equipment unique to the AP-600 design have been supported by R&D results especially for the passive features employed in the basic design.

Westinghouse is currently working on the detailed design in a First-Of-A-Kind Engineering (FOAKE) project. Detailed design for a standard AP-600 is expected to be ready for construction within the next 2 years. This would make the AP-600 available for construction as a standard plant on commercial basis with no further development needed before the end of this century.

In the conceptual design phase a research and development program was executed to confirm the passive safety features of the AP-600 system. This program included tests for the passive containment cooling system (PCS), the passive residual heat removal system (PRHRS) and a full size canned motor pump modified for higher inertia. In order to support the SSAR and give utilities added confidence in the system the second phase of R&D work was diversified in three main areas. These three areas are the PCS, the passive core cooling system (PXS), and the components design verification programs.

The PCS program encompassed testing of heat transfer from heated plates, an integrated PCS test of the heat removal capability using a steel pressure vessel of 0.9 m diameter and 7.3 m height, a large scale integrated PCS test of the heat transfer in a containment model in 1/8 scale, taking into account also internal structures and volumes to establish internal circulation patterns for air and steam mixture, testing of a 14° section of the containment in 1/6th scale to assess the air flow path resistance of air inlet and exhaust, testing of the effectiveness of water distribution on the dome and upper sidewall in 1/8th section full sized containment dome model, and tests (on 1/100 and 1/30-scale) at the wind tunnel facility of the University of Western Ontario to check the effect of wind speed and direction on the natural convection draft within the containment annulus.

The PXS program encompassed testing of the thermal performance of the PRHR heat exchanger in the IRWST and the mixing behaviour in the tank, testing of the ADS in a full-size simulation with one of the two flowpaths from the pressurizer at ENEA's VAPORE Test Facility in Casaccia, Italy, and testing of the capability of the PXS system check valves to open at low pressure differential conditions, testing of the function of the core makeup tank in a facility with a ~1/100 volume scale CMT, simulating CMT operating condition over a wide range to verify the gravity drain mechanisms, including testing of the instrumentation for CMT level indication, and a number of DNB tests at the Columbia University to extend the existing critical heat flux correlation for fuel assemblies at lower flow conditions.

A long-term cooling test has been conducted at Oregon State University to experimentally investigate the integral system and long-term cooling behaviour of the AP-600 nuclear steam supply system in a 1/4 scale reduced pressure (about 7.8 MPa) programme, including all safety systems and a proper representation of the geometry, and a full-height, full pressure integral system test at operating temperature and with full simulation of the AP-600 passive core cooling safety features, has been conducted at the SPES facility in Italy.

The component design verification programme has included testing at full operational speed of a full scale canned motor with a high inertia shaft, testing of a full scale control rod drive in a test stand with a simulated fixed detector to check possible interference problems, and an air flow test in a 1/9 scale model in clear plastic to investigate the effects of the removed bottom-oriented instrumentation in previous designs core flow patterns at different flow rates. Final testing of the check valves has been carried out as in-situ tests at the Farley nuclear power plant, at actual plant operation conditions,and an air flow test on the SG channel head and on the RCP, and tests to establish the water flow hydraulic performance of the RCP have been performed.

The licensing procedure in the US has been greatly simplified. A combined construction permit and operating license for a standard design plant has been agreed by the NRC. Title 10 of the CFR part 52 has been incorporated in a strategic plan by the nuclear power oversight committee allowing for early public participation in the design certification, site permit, and combined license issuance to avoid later delays. This licensing process consist of six parallel activities of which the first part is the submission of the SSAR. The SSAR consists of the safety aspects of the NSSS, balance of plant features and their safety impact. The SSAR was submitted to the NRC on June 26, 1992. The second part consists of the inspections, tests, analysis and acceptance criteria (ITAACs). Compliance with the ITAACs would demonstrate that public safety interests and investment are sound. The third part is submission of the Probabilistic Risk Assessment (PRA) report. After reviewing the SSAR the NRC issues a safety evaluation report (SER), followed by a final design approval (FDA) when the report has been fully approved. The design certification is a subsequent rule making processing; such certification would be valid for 15 years. The next and final milestone (in the US) is the application for (and issue of) a site specific, combined license by the utility wishing to construct and operate the plant on either a pre-licensed site or a new one.

References

No information provided.

5.3 HSBWR, HITACHI, JAPAN

5.3.1 Introduction

The HSBWR (Hitachi Simplified BWR) being developed by Hitachi is a design concept for a Boiling Water Reactor (BWR) in the small and medium size range. This concept aims at establishing a capacity series of BWRs up to a capacity of 600 MWe. The HSBWR adopts natural circulation of primary coolant and passive safety systems to improve the economy, maintainability, and reliability by simplification. The HSBWR is characterized by:

(a) system simplification by adopting natural circulation for coolant recirculation,

(b) a reactor building standardization with high seismic resistance,

(c) a high safety standards with an automatic depressurization system (ADS) and accumulators at low pressure for emergency core cooling systems (ECCSs) and an outer pool for decay heat removal, and

(d) a short construction period by adoption of a steel structure primary containment vessel.

The main features of the HSBWR with respect to these characteristics are delineated below:

1. The fuel assemblies are short compared to those in existing BWR plants, with a heated length of 3.1 m and a total length of 3.7 m, to avoid seismic resonance between the core and the reactor building. The latter may be constructed on soft to firm ground, which even makes it possible to construct the standardized plant underground with a high level of protection, if necessary;
2. The volumetric power density of the core is low, only 34.2 kW/l, and the operation period between refuellings amounts to 23 months;
3. The reactor internals arrangement is simple, without forced recirculation systems (i.e. the reactor operates with natural circulation of the reactor coolant) and steam separators;
4. There will be no core uncovery in any loss-of-coolant accident (LOCA) situation with coolant left in the reactor pressure vessel (RPV) and coolant injected by the actuation of the steam driven reactor core isolation cooling (RCIC) systems, the automatic depressurization system (ADS) and the accumulators at a relatively low pressure of 0.5 - 1.0 MPa;
5. The decay heat is absorbed in the suppression pool for one day after accident initiation, and heat conduction through the steel-walled primary containment vessel (PCV) to the surrounding outer pool provides heat removal from the containment for three days, giving operators ample time for recovery actions;
6. The flooding of the RPV by coolant fed from the accumulators through the RPV and direct cooling of its outer surface will help keeping the core covered with coolant and maintain core cooling, even in an RPV bottom break accident;
7. Depressurization by the ADS and injection of borated water from the accumulator at a pressure of 1 - 1.5 MPa to decrease reactivity and shut down the reactor in an anticipated transient without scram (ATWS) situation;
8. The primary containment vessel, reactor building and turbine building are standardized and compact, and the same plant layout will apply at any reactor site; and
9. A shortened construction period of 32 - 36 months (depending on the site conditions) including pre-operation and start-up tests will help ensure the economic competitivity.

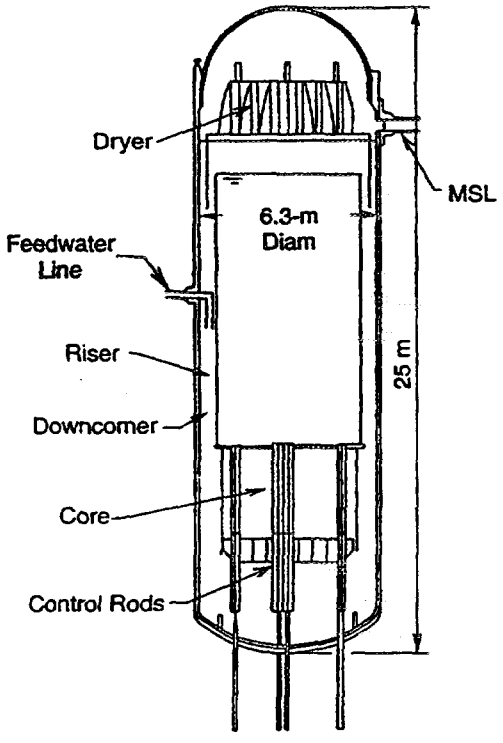


Figure 5.3-1 The HSBWR reactor vessel arrangement

5.3.2 Description of the nuclear systems

5.3.2.1 Primary circuit and its main characteristics

The schematic of the reactor pressure vessel (RPV) and the system configuration are shown in Figures 5.3-1 and 5.3-2, respectively.

Natural circulation is used for coolant circulation, i.e., pumped recirculation systems are eliminated, in order to reduce the number of components driven by external force and thereby improve the reliability and maintainability. A riser of 9 m height is installed above the core in order to increase the driving force for natural circulation. The 3.7 m length of the fuel assemblies, with an active length of 3.1 m, is determined by the objective of avoiding seismic resonance between the fuel bundles and the reactor building, independent of the conditions of the ground (soft to firm ground) on which it is constructed. This makes it possible to standardize the reactor building and plant layout without any connection to ground firmness.

Elimination of the steam separators results in a reduced flow resistance in the primary circuit, and the natural circulation flow rate becomes higher. The steam passes from the core and up the riser of 9 m height in 6 seconds, and the passage through the long steam dome takes 8 seconds. The long passing time decreases the strength of gamma-radiation from nitrogen N-16 and reduces the need for gamma shields in the turbine building.

The main coolant lines consist of two main steam lines of 700 mm pipe diameter with two main steam isolation valves (MSIVs), and two feedwater lines of 400 mm pipe diameter with isolation valves

5.3.2.2 Reactor core and fuel design

The power density of the reactor core is lower in a natural circulation reactor than in a forced circulation reactor, but the lower power density allows a longer continuous operation. The short heated length of the fuel and the low power density provide good thermal-hydraulic characteristics. With the 8x8 type fuel assembly selected, the power density is 34.2 kW/l; the number of fuel assemblies is 708, and the

equivalent core diameter is 4.65 m. The uranium enrichment of the refuelling batch, at equilibrium, is 3.6%, and the average fuel burn-up is 39 GWd/t in the event of 23 months operation cycles.

5.3.2.3 Fuel handling and transfer systems

The fuel handling and refuelling operations are quite similar to those in current BWR designs. After the containment dome and the reactor vessel head have has removed, the reactor pool above the vessel opening is filled with water to provide radiation shielding. Then the reactor internals are lifted out of the vessel, using the overhead crane, and placed in storage positions in the pool. The procedures are simplified, however, because the internals to be treated are only the steam dryers due to the elimination of the steam separators. The handling of the fuel is made by means of a traditional refuelling machine. Spent fuel assemblies are brought up to the reactor pool and transported to storage racks with a capacity of 5 operating cycles in an adjacent fuel storage pool.

5.3.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel is 23 m high and with an internal diameter of 6.3 m. The operating pressure is 7 MPa as in the ABWR, and the design pressure of the vessel is 8.7 MPa. The nozzles for the two main steam lines of 700 mm diameter and the two feedwater lines of 400 mm diameter are located in the upper portion of the vessel; no large diameter nozzles are situated below the top of the core.

Reactor internals

The reactor internals are simple. The core support structure and the arrangement of guide tubes for instrumentation systems and for the bottom-entry control rods are similar to those in other BWRs, but the natural, rather than forced, coolant recirculation results in somewhat reduced stresses. Above the core, there are more differences; there is a riser structure from the core outlet up to the steam dome to improve the natural circulation rate, and there are no steam separators.

Steam/water separation is achieved by gravity separation above the two-phase surface, but still steam dryers are installed at the top of the vessel for moisture separation, in order to ensure that the quality of the steam supplied to the turbine is acceptable.

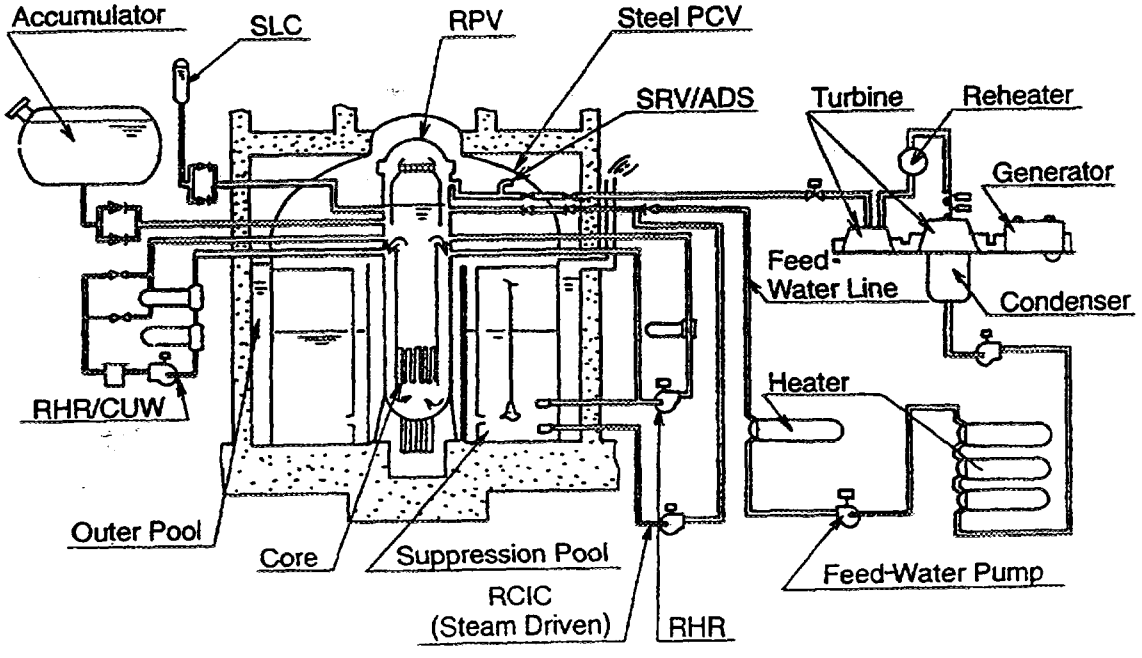


Figure 5.3-2 HSBWR - System configuration

Reactor recirculation pumps

The HSBWR is designed with natural circulation of the coolant and there are no recirculation pumps.

5.3.2.5 Reactor auxiliary systems

A reactor core isolation cooling (RCIC) system is installed to maintain the reactor water level by water injection from the condensate storage tank during isolation events. A reactor water clean-up (RWCU) system is installed to control the content of radioactive and corrosion products in the reactor water and to remove residual heat in case of failure of the residual heat removal (RHR) system. The number of RHR system trains is reduced to one by the system simplification with the accumulator and the common use of the reactor water clean-up system for residual heat removal.

Other auxiliary systems such as the spent fuel storage pool cooling system, are basically the same as in currently operating BWRs.

5.3.2.6 Operating characteristics

The load following between 100% and 50% is mainly performed by control rod manoeuvering since there are no recirculation flow control possibilities. The harmonized control system utilizing Artificial Intelligence (AI) and control of the extraction steam for the turbine are taken into account.

5.3.3 Description of turbine generator plant system

5.3.3.1 Turbine generator plant

The balance of plant of the HSBWR is mainly composed of two main steam lines, two feedwater lines, and the turbine systems, including one high pressure turbine and three low pressure turbines, etc.

The main turbine is a tandem compund, double flow turbine with 52" (1320 mm) blades on the last stage of the low pressure (LP) turbine units. The turbine, TCDF-52, is an adoption of the TC6F-52 which had already been developed for the 1350 MWe BWR (ABWR), in order to simplify the turbine island. A moisture separation and steam reheating system of the same type as in the ABWR plant is installed to improve the efficiency.

Bypass system is not shown and described in the text.

5.3.3.2 Condensate and feedwater systems

The condensate transfer pump is minimized by using the re-entry type. A high efficiency of about 33.4% is estimated under the condenser vacuum of 722 mm Hg. The feedwater system consists of the main feed pump, feedwater heaters, and piping in the same way as in the currently operating BWRs, even though the number of heaters is reduced from six to four per train, and the number of trains has decreased to one.

5.3.3.3 Auxiliary systems

The auxiliary systems are basically the same as in currently operating BWRs.

5.3.4 Instrumentation and control systems

5.3.4.1 Design concepts including control room

The concept of I&C and electrical systems in HSBWR are basically the same as those of the ABWR. However, the electrical load is much smaller since the recirculation system has been eliminated. Furthermore, I&C and electrical systems are much simplified, e.g. the capacity of the emergency diesel generators is reduced, due to the adoption of passive safety systems.

5.3.4.2 Reactor protection system and other safety systems

The reactor protection system and the other safety-related control systems are basically the same as those in current BWRs. The signals for the reactor protection system are generated in 2-out-of-4 coincidence logics. The digitalization of control panels and monitoring with CRT (cathode ray tube) displays are the main differences from the currently operating BWRs.

5.3.5 Electrical systems

5.3.5.1 Operational power supply systems

The concept of the ordinary power distribution is basically the same as that of the current BWRs. The power supply system capacity is greatly reduced, however, due to the elimination of the recirculation pumps. The number of DC power supply trains in the ordinary system is reduced to one.

5.4.5.2 Safety-related systems

The power supply for the safety-related system is reduced in accordance with the introduction of passive safety systems. One diesel generator unit of 1500 kW is provided to improve power supply security; it is not needed for emergency. The number of DC power supply trains in the safety-related system is two. The batteries have a capacity that is sufficient for 2 hours at rated load.

5.3.6 Safety concept

5.3.6.1 Safety requirements and design philosophy

The safety design philosphy are as follows:

- To increase reliability of systems and avoid system complexity by adoption of passive safety systems.
- To increase grace period with respect to severe accidents.

The safety requirements are that there shall be no core uncovery in the event of design basis accidents and containment heat removal shall be ensured for 3 days without external power or other external support functions. The first requirement is the same as for the ABWR, and the corresponding design requirement is that no large diameter piping shall be connected to the reactor pressure vessel below the top of the core has been taken into account in the vessel design. A relatively large capacity water accumulator is introduced for emergency core cooling (ECC) purposes instead of the systems with motor-driven ECC pumps in the currently operating BWRs. The second requirement is met by the introduction of a water-wall containment cooling system which takes advantage of the steel primary containment vessel structure.

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

The HSBWR configuration provides a high safety level with respect to postulated loss-of-coolant-accidents (LOCAs) by pipe breaks, because there are no large diameter pipes connected to the reactor pressure vessel below the top of the core. The safety systems have redundancy by the combination of active and passive system.

The steam driven reactor core isolation cooling (RCIC) system is provided for loss of all AC power (i.e. station blackout) situations and are also effective for small break LOCAs. The automatic depressurization system (ADS) and accumulators with emergency coolant at a low pressure (some 0.5 - 1 MPa) provide for short term emergency core cooling, instead of the high and low pressure pumped injection systems and emergency diesel generators in the current BWR designs. Elimination of emergency diesel generators and pumped injection systems simplifies the emergency core cooling systems (ECCSs) and provides for a high system reliability because of fewer components. The ADS flow area, initial pressure, and piping diameters of the accumulators are determined to realize no initial core heat-up and no core uncovery in any LOCA situation. The accumulators have the capacity to keep the whole core covered with coolant during the initial 24 hours after occurrence of an accident.

Core cooling after normal reactor shutdown and long-term core cooling after reactor scram are performed by a residual heat removal (RHR) system with injection pumps and heat exchangers. The RHR system has the capability to cool down the reactor to 52°C within 20 hours, and its capacity is sufficient also for heat removal during accidents. The accumulators and the suppression pool can remove and absorb heat for one day after reactor scram by themselves. Even when the RHR system is not available, coolant can be fed into the RPV by manually refilling the accumulators with attachable pumps, and heat removal from the RHR is performed by heat transfer from the suppression pool through the steel-walled primary containment vessel (PCV) to the outer pool. This heat removal from the PCV to the outer pool needs no additional systems. The outer pool has the capacity to maintain heat removal from the PCV to the outer pool for more than three days. Also, it is possible to feed water into the outer pool when it is requested.

5.3.6.3 Severe accidents (Beyond design basis accidents)

With respect to severe accidents, the containment cooling capability without any operator procedures is under investigation. The containment cooling by the outer pool (a kind of water wall type passive containment cooling system) is maintained for more than three days without any procedures.

5.3.7 Plant layout

5.3.7.1 Buildings and structures, including plot plan

The layout of the reactor island is shown schematically in Figure 5.3-3. The volume of the reactor building is about 50% of a current BWR building for the same rated capacity, which is realized by simplifying the components and systems as described above and by moving the spent fuel pool and control room to other buildings. All water pools are manufactured to the high level of welding technology and skill available in Japan, and steel structures are adopted for the primary containment vessel (PCV). Simplification of components and systems and adoption of steel structures in the PCV make it possible to achieve a shortened construction period which will be 32 - 36 months from the start of construction to commercial operation.

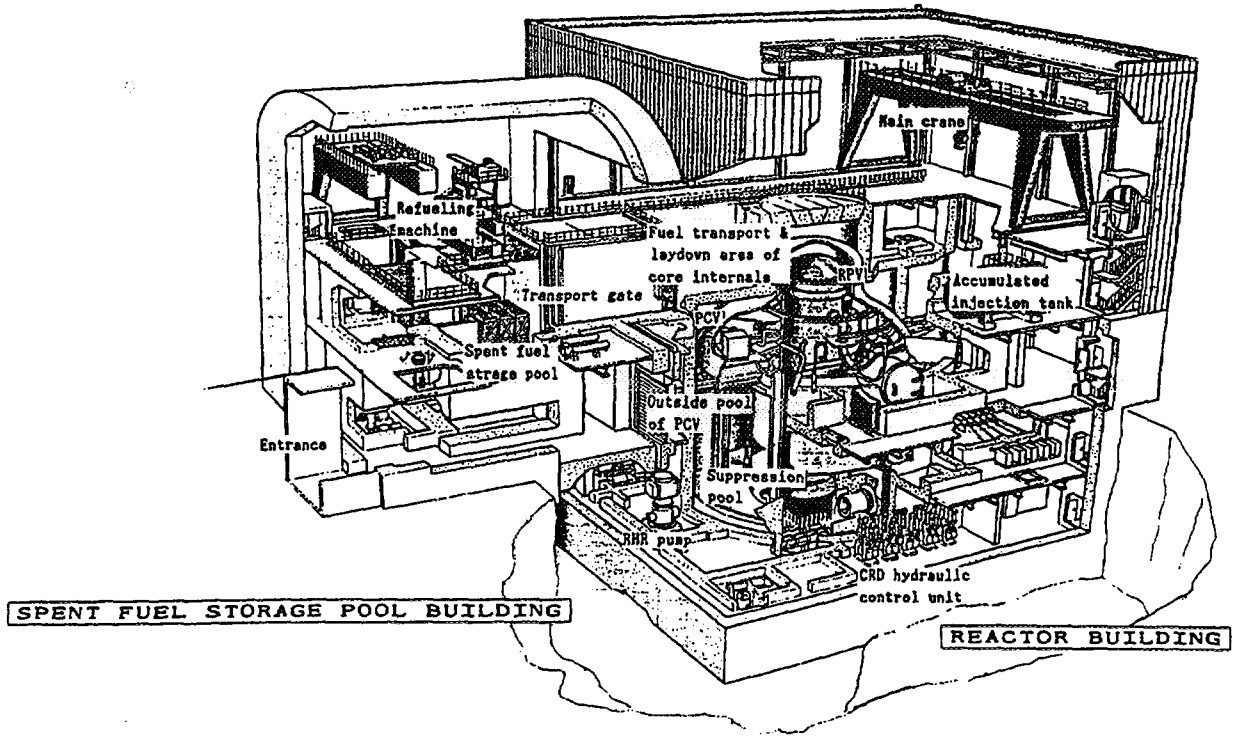


Figure 5.3-3 General layout arrangement of the HSBWR reactor

Design requirements

Earthquake: The seismic design is based on the Japanese guide S1 wave scaled to 0.35g for the firmest ground (with Vs+2000 m/s.

Aircraft crash, Explosion pressure wave, Internal hazards, Physical separation aspects, and Radiation protection aspects (accessibility, shielding, ventilation): No information provided

5.3.7.2 Reactor building

The reactor building encloses the containment and houses service systems for the reactor, such as handling equipment for fuel and main components, fuel pool, reactor water cleanup system and engineered safety systems.

In comparison with a current-day BWR of the same rated power, the building volume is decreased to 46% by the elimination of the recirculation pumps, ECCS pumps, and emergency diesel generators. The control room is separated and installed in another building to make the reactor building even more compact. The outer pool is located between the containment wall and the reactor building wall. The dimensions of the reactor building are 47 m width times 47 m depth times 47 m height. The seismic design is based on the Japanese guide S1 wave scaled to 0.35g for the firmest ground (with Vs+2000 m/s), and the standardization of the reactor building for any ground firmness is confirmed.

5.3.7.3 Containment

The containment is a free-standing steel containment vessel with a geometry that corresponds to the pressure suppression type containment of the current BWRs. The dimensions of the containment are 23 m diameter times 35 m height, and the building volume is about 70% of that of a current BWR of the same rated power.

Containment integrity, and corium retention

No information provided

5.3.7.4 Turbine building

The turbine building is directly adjacent to the reactor building enclosing the BOP systems such as the HP and LP turbines, the generator, the main condenser, and the feedwater pump. The dimensions of the turbine building are 47 m width times 58 m depth times 45 m height, and the volume is about 56% of that of a current BWR of the same rated power due to the system simplifications.

5.3.7.5 Other buildings

Each plant (unit) will be provided with an electrical/control building, a service building, and a fuel storage building, and there will be one waste building per four units at a power station. The structures of these buildings will be quite conventional.

5.3.8 Technical data

General plant data		
Power plant output, gross		MWe
Power plant output, net	600	MWe
Reactor thermal output	1 800	MWt
Power plant efficiency, net	30	%
Cooling water temperature	19	°C
Nuclear steam supply system		
Number of coolant loops	1	
Primary circuit volume		m^3
Steam flow rate at nominal conditions	803.3	kg/s
Feedwater flow rate at nominal conditions	899.2	kg/s
Reactor coolant system		
Primary coolant flow rate	4 944	kg/s
Reactor operating pressure	7.0	MPa
Steam temperature/pressure	289/7.0	°C/MPa
Feedwater temperature	185	°C
Core coolant inlet temperature		°C
Core coolant outlet temperature		°C
Mean temperature rise across core		°C
Reactor core		
Active core height	3.1	m
Equivalent core diameter	4.65	m
Heat transfer surface in the core		m^2
Average linear heat rate		kW/m
Fuel weight		t U
Average fuel power density		kW/kg U
Average core power density	34.2	kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		
Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	3 700	mm
Rod array	8×8 in square lattice	
Number of fuel assemblies	708	
Number of fuel rods/assembly	60	
Number of spacers		
Enrichment (range) of first core, average		Wt%
Enrichment of reload fuel at equilibrium core	3.6	*Wt%*
Operating cycle length (fuel cycle length)	23	months
Average discharge burnup of fuel	39 000	MWd/*t*
Cladding tube material	Zircaloy	
Cladding tube wall thickness	0.86	mm
Outer diameter of fuel rods		mm
Fuel channel/box; material		
Overall weight of assembly, including box		kg
Uranium weight/assembly		kg
Active length of fuel rods	3 100	mm
Burnable absorber, strategy/material	Gd_2O_3 mixed with fuel	
Number of control rods	169	
Absorber material	B_4C/Hf	
Drive mechanism	electro-mechanical	
Positioning rate		mm/*s*
Soluble neutron absorber	not used	
Reactor pressure vessel		
Inner diameter of cylindrical shell	6 300	mm
Wall thickness of cylindrical shell		mm
Total height, inside	24 550	mm
Base material: cylindrical shell	low-alloy carbon steel	
RPV head		
lining		
Design pressure/temperature	8.79/302	MPa/°C
Transport weight (lower part) [include. head]		t
RPV head		t

Reactor recirculation pump		Not applicable	
Type			
Number			
Design pressure/temperature			MPa/°C
Design mass flow rate (at operating conditions)			kg/s
Pump head			MPa
Rated power of pump motor (nominal flow rate)			kW
Pump casing material			
Pump speed (at rated conditions)			rpm
Pump inertia			kg m^2
Primary containment			
Type		Single wall, Pressure-suppression	
Overall form (spherical/cyl.)		cylindrical, steel	
Dimensions (diameter/height):		23/35	m
Design pressure/temperature		550/170	kPa/°C
Design leakage rate		0.5	vol%/day
Is secondary containment provided?		yes, surrounding reactor building	
Reactor auxiliary systems			
Reactor water cleanup,	capacity	22	kg/s
	filter type	deep-bed ?	
Residual heat removal,	at high pressure	250	kg/s
	at low pressure (100 °C)	6.3	MW
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s
Power supply systems			
Main transformer,	rated voltage		kV
	rated capacity		MVA
Plant transformers,	rated voltage		kV
	rated capacity		MVA
Start-up transformer	rated voltage		kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)			
Number of low voltage busbar systems			
Standby diesel generating units: number		2	
	rated power	1.5	MW
Number of diesel-backed busbar systems			
Voltage level of these			V AC
Number of DC distributions 0			
Voltage level of these			V DC
Number of battery-backed busbar systems			
Voltage level of these			V AC
Turbine plant			
Number of turbines per reactor		1	
Type of turbine(s)	TCDF-52	tandem compound, double flow	
Number of turbine sections per unit (e.g. HP/LP/LP)		1 HP/3 LP	
Turbine speed		1 500	rpm
Overall length of turbine unit		22	m
Overall width of turbine unit			m
HP inlet pressure/temperature		6.92/	MPa/°C
Generator			
Type		4-pole, 3-phase, turbo-generator	
Rated power		670	MVA
Active power			MW
Voltage		20	kV
Frequency		60	Hz
Total generator mass, including exciter			t
Overall length of generator			m
Condenser			
Type		shell type	
Number of tubes			
Heat transfer area		41290	m^2
Cooling water flow rate		41.9	m^3/s
Cooling water temperature		19	°C
Condenser pressure [722 mm Hg]		96.2	kPa
Condensate pumps			
Number		3 Hp & 3 LP	
Flow rate	[1850 m^3/h]	500	kg/s
Pump head	HP/LP [190/150 m WG]	1.86/1.47	MPa
Temperature			°C

Pump speed			rpm
Condensate clean-up system			
Full flow/part flow		full condensate flow	
Filter type		rod type	
Feedwater tank			
Volume			m^3
Pressure/temperature			MPa/ °C
Feedwater pumps			
Number			
Flow rate			kg/s
Pump head			MPa
Feed pump power			MW
Feedwater temperature (final)		185.2	°C
Pump speed			rpm
Condensate and feedwater heaters			
Number of heating stages,	low pressure	4	
	high pressure	1	
	feedwater tank	NA	

5.3.9 Project status and planned schedule

The HSBWR design is still at the conceptual design stage and licensing reviews have not yet been started.

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5.4 MS-600, MITSUBISHI, JAPAN

5.4.1 Introduction

Next generation light water reactor designs are currently being developed all over the world, and many concepts include application of passive safety features, i.e. features that utilize natural physical phenomena as the driving force. Passive components and systems are often being considered for adoption to replace conventional safety systems using active components, as a way of further improving nuclear power plant safety. Major targets in this context are:

- Simplification and improved reliability of safety systems, as well as cost reductions;
- Reduced reliance on the operators, and elimination of human factor errors which sometimes negatively affect operations during accidents; and
- Providing sufficient time margins to enable the operators to cope with accidents.

After analyses of applying passive safety concepts at the system level, and taking into detailed consideration potential limitations of the passive features, Mitsubishi has decided to adopt a concept of hybrid safety systems, which it anticipates will be adopted by many next generation plant designs.

The hybrid safety systems studied in the first place, are intended for use in a 2-loop PWR plant with an output in the 600 MWe class designated the MS-600 (600 MWe Mitsubishi Simplified PWR). The main feature of the MS-600 is in its safety systems.

5.4.2 Description of the nuclear systems

5.4.2.1 Primary circuit and its main characteristics

The configuration of the primary system is shown in Figure 5.4-1. The main characteristics of the primary system are as follows:

- the steam generators are of horizontal type and part of the passive safety design. Horizontal steam generators have a number of advantages, including freedom from sludge build-up on tube plates, and a better capability for withstanding possible seismic events.
- The reactor coolant pumps are of a high-efficiency type. The No.1 seal is a ceramic monoblock seal and high temperature secondary seals are being developed to improve both the seal performance and the service life time, as well as to give greated endurance during a loss-of-seal-cooling event which will occur when all AC power supply is lost.

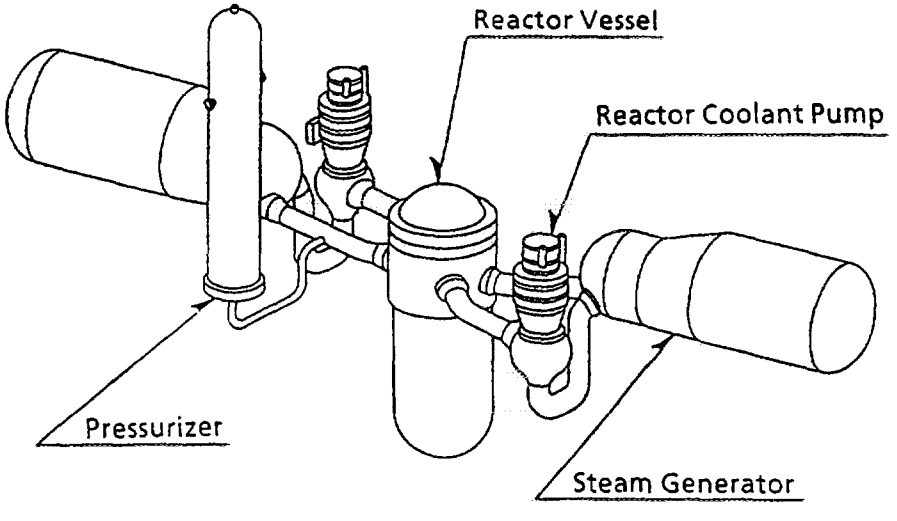


Figure 5.4-1 MS-600 - Primary system configuration

5.4.2.2 Reactor core and fuel design

The reactor core design of the MS-600 is essentially based on the experience gained with currently operating plants. The core is composed of 145 fuel assemblies that are of identical mechanical design. Each fuel assembly consists of 264 fuel rods, 24 guide thimbles for the control rod cluster and one guide thimble for the in-core neutron detector. The hole for the top entry in-core detector is located at top of the in-core detector guide thimble, and the bottom of the thimble has a small hole for cooling purposes. The fuel rods are arranged in a 17×17 square lattice array, and their active length is 3 648 mm (12 ft.).

The power density of the reactor core is low in comparison with previous 600 MWe designs. This results in lower fuel enrichments, longer achievable fuel cycles, and increased operating and safety margins.

5.4.2.3 Fuel handling and transfer systems

The fuel handling and storage facilities provide the means for safe and reliable handling and storage of fuel assemblies and control components from the time new fuel is delivered to the plant through refuelling and storage to final dispatch of the spent fuel.

The fuel handling and storage facilities are located in the reactor containment and fuel handling building.

The reactor cavity, located immediately above the reactor, provides an area inside the containment where the upper core internals and the lower core internals can be stored during refuelling. The cavity is filled with boric acid solution at the time of refuelling. A manipulator crane is installed in the reactor cavity.

The fuel storage pit and a cask loading pit are located inside the containment.

5.4.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel is cylindrical, with a hemispherical bottom and upper head. The upper head is removable, and is connected to the vessel flange by stud bolts. The flange sealing is provided by O-rings.

The vessel contains the core, reactor internals, control rods, movable detectors, thermocouples and other parts associated with the reactor core. The control rods are operated by sealed drive mechanisms mounted on the upper head. The movable detectors are inserted through adapters mounted on the upper head. There are no adapters and no instrumentation connections in the bottom of the reactor pressure vessel.

The vessel has two inlet and two outlet nozzles. Coolant enters the vessel through the inlet nozzles and flows down along the inner wall of the vessel, turns at the bottom, and flows up through the core and up to the outlet nozzles. The structural portion of the vessel is made of low-alloy carbon steel with a weld deposit stainless steel cladding on all surfaces exposed to the reactor coolant.

The reactor pressure vessel is designed for a life time of 60 years, with a total neutron fluence not exceeding 10^{19} nvt. This has been achieved by adoption of a rather large water gap and a radial reflector of ring block type around the core.

Reactor internals

The upper core internals consist of the control rod guide assemblies, the upper core support plate, the upper core plate, the upper support columns and the guide tubes for the movable in-core detectors. The support columns are hollow tubes and are used as guides for the thermocouples and movable detectors. Six movable detector guide thimble tubes are bunched together, and taken out through one common reactor vessel head instrumentation penetration. The instrumentation penetration is of the same type as that used for the thermocouples.

The lower core internals consist of the core barrel, the lower core plate and a radial reflector. The structure in the space between the lower core plate and the vessel bottom head has been simplified as a result ofusing top-mounted instrumentation, and a vortex suppression device is installed to maintain a stable coolant flow. The upper and lower internals are clamped together by means of a flexible ring spring at the vessel flange. The coolant flows through the lower core plate and the flow is distributed uniformly through each fuel assembly by the core plate orifices.

The space between the core periphery and the cylindrical inner surface of the core barrel is filled by a ring block reflector made of stainless steel to reduce the fast neutron leakage and to flatten the power distribution.

Steam generators

Steam generators of horizontal type have been chosen in order to avoid a possible siphon break in the natural circulation mode due to accumulation of non-condensable gases in the steam generator tubes.

The horizontal steam generator is of a new design which, compared to a conventional vertical type steam generator, is less likely to have sludge accumulation on the tube plates during normal operation. The horizontal steam generator has a much lower height and it will therefore be more resistant to earthquakes.

Pressurizer

The pressurizer is of conventional design. The pressure of the reactor coolant system (RCS) is controlled by the pressurizer; in the pressurizer an equilibrium between the liquid phase and the steam phase is maintained by heaters and sprays. The pressure changes due to the expansion and the shrinkage of the reactor coolant upon load changes are absorbed by condensation of steam, using the pressurizer sprays, or by generation of steam, using the pressurizer heaters. The pressurizer has two spring type safety valves and eight motor-operated valves (primary valves of the automatic depressurization system). If the pressure in the pressurizer becomes too high, the safety valves discharge steam to the gravity injection pit in which the steam is condensed and cooled by mixing with the water in the pit. The primary automatic deprssurization valves also discharge the steam into the gravity injection pit where it condenses. The primary automatic depressurization valves are actuated if a LOCA should occur, in order to depressurize the RCS to a pressure at which the passive gravity injection system can operate.

Reactor coolant pumps

The reactor coolant pumps are of a high-efficiency type. The No. 1 seal is a ceramic monoblock seal and high temperature secondary seals are being developed to improve both the seal performance and the service life time, as well as to give greated endurance during a loss-of-seal-cooling event which will occur when all AC power supply is lost.

Main coolant lines

The main coolant lines will be made in forged ferritic steel with a cladding of austenitic steel. The Leak-Before-Break concept will be applied. A high quality in design, construction and surveillance enables preclusion of the current pipe break assumptions; consequently a catastrophic failure of a main cooling line is ruled out as regards its possible mechanical effects. However, a mass flow equivalent to a double area break of a main coolant line is still assumed for the design of the emergency core cooling system.

5.4.2.5 Reactor auxiliary systems

Chemical and volume control system

The principal functions of the Chemical and volume control system (CVCS) are to control the chemistry, purity and inventory of the water of the reactor coolant system (RCS), and to provide seal water injection, as delineated below:

- Maintain programmed water level in the pressurizer, i.e., maintain required water inventory in the RCS;
- Maintain seal water injection to the reactor coolant pumps;

- Control reactor coolant water chemistry conditions, activity level, and soluble neutron absorber concentration;
- Provide means for filling, draining and pressure testing of the RCS; and
- Provide reactor coolant purification during a cold or a refuelling shutdown.

In addition to the above functions, the CVCS provides makeup water to the RCS in the event of very small LOCAs without actuation of the passive safety systems. Therefore, the charging pumps are designed to have the function of safety injection also. The safeguards function of the charging pumps is explained in Section 5.4.6.2.

The main function of the CVCS is realized by a combination of charging and letdown flows. Reactor coolant is discharged to the CVCS from the reactor coolant cold leg piping; it then flows through the shell side of a regenerative heat exchanger in which the temperature is reduced to below 150°C during normal plant operation. The pressure is reduced by passing the flow through one or more letdown orifices, and then the coolant flows through the tube side of the letdown heat exchanger where its temperature is reduced further to about 55°C. There are three orifices in parallel, and the letdown flow rate can be varied by selecting combinations of these orifices. The coolant normally flows through a mixed bed demineralizer, through a reactor coolant filter, and into the volume control tank through a spray nozzle in the gas space of the tank.

The bulk of the charging flow is pumped back to the RCS through the tube side of the regenerative heat exchanger. A flow path is also provided from the regenerative heat exchanger outlet to the pressurizer spray line which is used to cool the vapour space of the pressurizer at the end of plant cooldown. A portion of the charging flow is directed to the reactor coolant pumps through a seal injection heat exchanger and filter.

5.4.2.6 Operating characteristics

The MS-600 is designed for being automatically operated between 0 and 100% of rated power. In the lower power range, below 15% power, the rod control system and the feedwater control system are switched to the low power control mode.

The average reactor coolant temperature is normally controlled at the programmed reference temperature which varies with the turbine load. When the reactor is controlled in the daily load following mode, a variable temperature programme is used, taking advantage of the moderator temperature feedback effect, to reduce water processing and enhance plant capability. The variable temperature program automatically determines the reference temperature and returns it back to the normal reference temperature when the load change stops.

The control systems are designed to provide the MS-600 with a high load following capability. The reactor can follow the load demands listed below without reactor trip:

- ± 10% step load change within a range of 15 and 100% of rated power;
- ± 5%/min ramp load change within a range of 15 and 100% of rated power; and load reduction.

The reactor is also capable of load following and participating in grid frequency control as follows:

- 100-50-100% step load follow operation (14 hours at full load, 8 hours at 50%, and one hour load changes); and
- ± 5% grid frequency control (both automatic frequency control and governor free control modes).

5.4.3 Description of turbine generator plant system

5.4.3.1 Turbine generator plant

The high pressure turbine is a double-flow unit with a rateau control stage followed by reaction blading at each end of the unit. The steam enters the high pressure turbine through two throttle valve-steam chest assemblies, one located on each side.

Each low pressure turbine is a double-flow unit which employs reaction blading. Steam enters at the centre of the unit, flows through the blading to an exhaust opening at each end, and then downward to the condenser.

A low presure turbine with 44" (1120 mm) last stage blades is used to give a high electrical output and increased thermal efficiency. High performance blades, multi-seal fins and a high performance exhaust flow guide are used to improve the turbine efficiency.

An improved moisture remover is adopted for erosion protection.

The generator is a four-pole turbogenerator, designed for continuous operation with hydrogen as cooling medium for the rotor and for the stator windings.

The exciter for the turbine generator is a brushless exciter with a simplified rotating rectifier which uses high performance diodes.

The moisture separator and reheater employs a high performance separator to handle the increased inlet steam flow and is designed to be compact.

5.4.3.2 Condensate and feedwater systems

Condensate is drawn from the condenser hotwells by three half-capacity, motor-driven condensate pumps. The condenser hotwells are interconnected by an equalizing header. Therefore, even if one of the three pumps is out of service, full load can be maintained.

The condensate pumps discharge into a common header. The condensate then passes through the gland steam condenser and the condensate demineralizer to the condensate booster pumps. These pumps discharge into two parallel strings of low pressure heaters, each string consisting of four low pressure extraction feedwater heaters. Downstream these strings the condensate goes to the deaerator and storage tank (the feedwater tank).

The condensate from the deaerator storage tank is routed to three motor-driven feedwater pumps, each of 50 percent capacity. The feedwater pumps are arranged in parallel, and the discharge from the pumps is directed through two parallel strings of high pressure feedwater heaters (in six stages) to a common header and from there to the individual feeding lines for the steam generators.

The feedwater flow rate to each steam generator is regulated through individual feedwater control valves positioning by a three-element feedwater control system associated with each steam generator. The three elements are the feedwater level in the steam generator, the steam flow rate and the feedwater flow rate.

The main condenser is actually two condensers, each connected to one low pressure turbine exhaust. Each condenser is horizontal, radial-flow, single-pass, and surface cooled, with divided water boxes and a turbine exhaust steam inlet at the top. The condenser is firmly fixed to the basement floor, and is connected to the turbine exhaust with a rubber bellows connection. The condenser is designed to maintain a pressure of 5 kPa (0.05 bar) at the rated circulating water temperature of 21°C. The condenser tubes are made of titanium.

The feedwater flow rate is regulated by three-element control valves. The feedwater pumps have steadily rising head characteristics and are capable of operating smoothly from minimum to maximum flow, without hunting, vibration or other undesirable effects.

5.4.3.3 Auxiliary systems

The unit has two circulating water pumps. Each pump serves a divided water box section of each single-pass condenser. There are separate discharge lines from each half-section of the condensers to each discharge channel.

Each of the pumps is a half-capacity, vertical, dry pit, single stage pump, connected by rigid coupling to a vertical, 200 rpm, 3-phase, 50 Hz, 6.6 kV induction motor.

5.4.4 Instrumentation and control systems

5.4.4.1 Design concepts including control room

I&C system architecture

The I&C system of the MS-600 is a micro-computer based digital control system, divided into the following four levels, each classified in accordance with its function:

Station level:	Administration and management function for the whole station
Unit level:	Plant monitoring and control functions for the unit
System level:	Protection and control functions for each I&C system interfacing with the local level
Local level:	Sensing and monitoring, protection and control functions for each piece of local equipment interfacing with the system level.

The design features of this system architecture are the following:

- Application of digital technology for all I&C systems;
- Distributed architecture with complete redundancy;
- Fully computerized main control room; and
- Fully multiplexed signal transmission network.

Main control room

The main control room is of an advanced type incorporating state-of-the-art technology with compact consoles using CRTs (Cathode Ray Tubes) and FDPs (Flat Display Panels), and overview panels with LDPs (Large Display Panels) which continuously display overall plant information and alarm status.

The display screens provide support to the operators by processing various kinds of information, and operator control actions are taken via control panels of touch screen type, i.e., monitoring and control functions are integrated into one screen. Computer systems can provide automatic plant status checking, automatic displays, dynamically prioritized alarms.

5.4.4.2 Reactor protection system and other safety systems

To improve safety and attain a lower probability of core damage (Core Damage Frequency or CDF in probabilistic safety assessment (PSA) terminology), the reactor protection system (RPS) and the actuation systems for engineered safeguards (ESF) should have sufficient redundancy and functional diversity in each system. The main design features of the safety-grade digital I&C systems are:

- Complete redundancy and functional diversity
  - Four channel reactor protection system and two-train ESF actuation system;
  - Two trip subsystems per channel;
  - Two data communication susbsystems

- Fail-to-safe design
  - Dynamic trip logic;
  - Self-diagnostic function;
- Testability
  - Fully automatic testing;
- Maintainability
  - Self-diagnostic function;
  - Modular architecture;
  - Visual software maintenance tool.

5.4.5 Electrical systems

The configuration of the electrical systems is shown on the single line diagram of Figure 5.4-2.

5.4.5.1 Operational power supply systems

The operational power supply systems of the MS-600 consist of the following two groups of systems:

- The operational power supply systems as in currently operating PWR plants, comprising:
  - Four 6.6 kV AC buses;
  - Four 440 V AC buses;
  - Two DC power supply systems (1× 6000 Ah, 1×3500 Ah).
- A permanent power supply system (a dedicated non-safety system) which shall supply power to preserve the plant in the event of a loss of external power supply:
  - Two 6.6 kV AC buses;
  - Four 440 V AC buses;
  - One diesel generator.

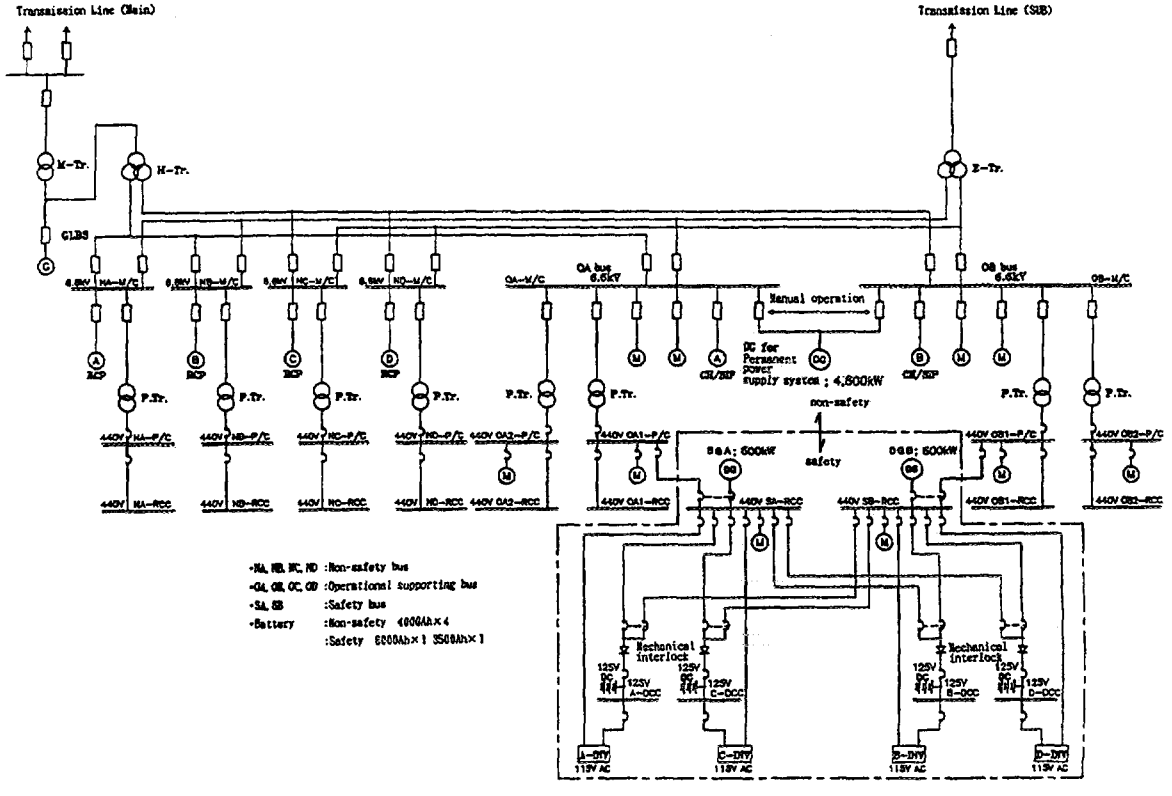


Figure 5.4-2 MS-600 - Single line diagram

5.4.5.2 Safety-related systems

The safety-related power supply systems of the MS-600 are the DC power supply systems for the four train of DC motor-operated valves. The safety-related DC power supply systems are backed up by diesel generators, taking into consideration a long term loss of external power supply after an accident, and to reduce the capacity of the batteries. Hence, the MS-600 configuration includes the following safety-related power supply systems:

- Two 440 V AC buses;
- Two diesel generators;
- Four safety DC power supply systems (4×4000 Ah).

5.4.6 Safety concept

5.4.6.1 Safety requirements and design philosophy

Deterministic design basis

The safety design of the MS-600 is based on the deterministic rules applied for the safety analyses of current PWR designs, such as the single failure criterion and the safety evaluation limits to be satisfied for various kinds of abnormal and accidental events. No additional initiating events are assumed, compared with currently operating PWRs.

Risk reduction

Mitsubishi has also concluded that it is important to apply probabilistic safety assessment (PSA) methodology as much as possible from the early stage of the basic design in order to cover multiple accident conditions beyond the design expectations and to achieve a safety level exceeding that of currently operating plants. A very low core damage frequency (CDF) value has been attained by the diversity of safety systems, the highly reliable passive systems and the elimination of human actions during accidents.

A risk reduction has been achieved as can be seen from the much lowered core damage frequency compared to that of currently operating PWRs (more than one order of magnitude lower) and also by the incorporation of mitigation features for severe accidents (Section 5.4.6.3).

External and internal hazards

External hazards have not been considered yet, since these are site-dependent.

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

Safety systems configuration

The innovative features of the MS-600 are the hybrid safety systems (an optimum combination of active and passive safety systems) and horizontal steam generators. With this combination, the functions of the passive safety systems have been expanded, and the active safety components have been greatly simplified. The entire safety system has been simplified so that improved safety, higher reliability and better economy can be obtained.

The development of the new concept of the hybrid safety systems started with analysis and assessment of advantages and disadvantages of each type of safety system.

One of the advantages of the active safety systems is their effectiveness in quickly terminating an accident and in preventing the expansion of accidents. The active safety systems also provide operational flexibility under different accident conditions, and they can be used in the best way and in the best combination according to the operator's judgement. Their disadvantages are the possibilities of aggravating an accident by operational errors or misunderstandings on the part of the operators and the complexity of the systems, which require a series of active components, including sources of power, to operate correctly.

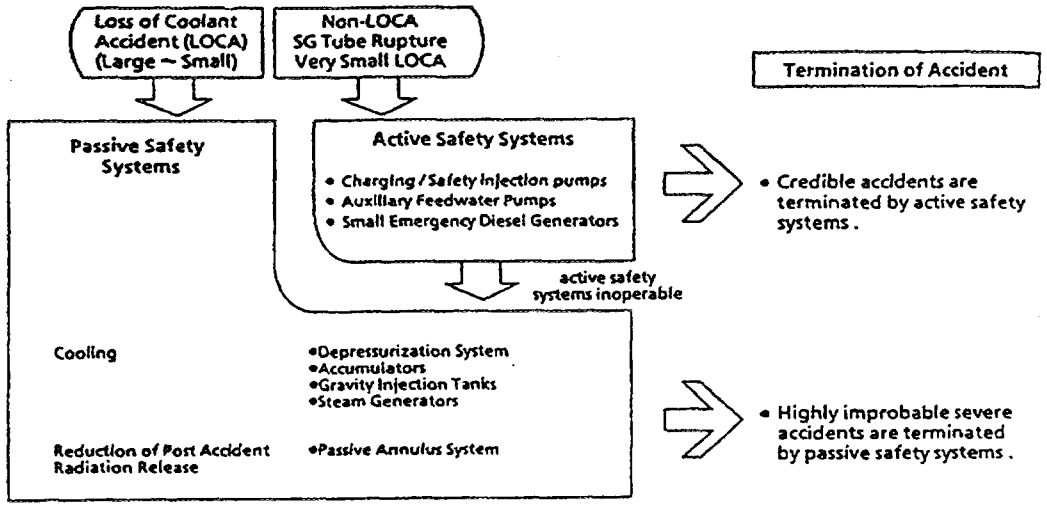


Figure 5.4-3 MS-600 - The basic hybrid safety systems concept

Passive safety systems are simple, highly reliable, and allow the designer to eliminate human errors that sometimes make accident situation worse, because they are composed totally of passive components that use natural forces and do not require operator actions or a series of operations of peripheral components. On the other hand, when passive safety systems are used, terminating an accident takes longer time, and the operational procedure is fixed, so that, in some cases, even a small accident eventually results in flooding the inside of the containment, which in turn requires a tremendous recovery effort.

The basic concept of the hybrid safety system (Figure 5.4-3) was developed based on these considerations.

The safety systems used to terminate an accident situation depend on the accident event; passive safety systems handle loss-of-coolant-accidents (LOCAs) which are considered highly unlikely accidents, whereas more likely accidents, such as power supply failures (blackouts), failures of secondary system piping, or steam generator tube ruptures (SGTRs) are handled by active safety systems. Breaks of primary system piping with a diameter equal to or less than 25 mm (1 in.), very small LOCAs, are also dealt with by active safety systems. The passive systems act as back-ups to prevent core damage if the active safety systems do not operate correctly due to operational errors or some other reason. Such an optimum combination of systems can improve safety while maintaining the advantages inherent in the present systems.

Passive safety systems

The MS-600 passive safety systems make maximum use of natural phenomena (natural circulation, gravity, gas pressure, etc.). These systems consist of the automatic depressurization system, the advanced accumulators, the gravity injection pit and the horizontal steam generators with feedwater supply by gravity from a condensate storage tank (Figure 5.4-4). Specific design features of the passive safety systems are presented below.

Automatic depressurization systems. In order to make it possible to inject water by gravity into the reactor coolant system and the secondary side of the steam generators, automatic depressurization systems (ADSs) are installed in both primary and secondary systems. In a large break LOCA event, the RCS is depressurized rapidly to a pressure below the gravity head.

For small and intermediate sizes of breaks, however, it is necessary to depressurize the RCS so that the passive gravity injection system can work. This is done by the primary and secondary ADSs. The actuation logic actuates the primary ADS (PADS) at an early stage of an event such as a small LOCA, and this is followed by the actuation of the secondary ADS (SADS).

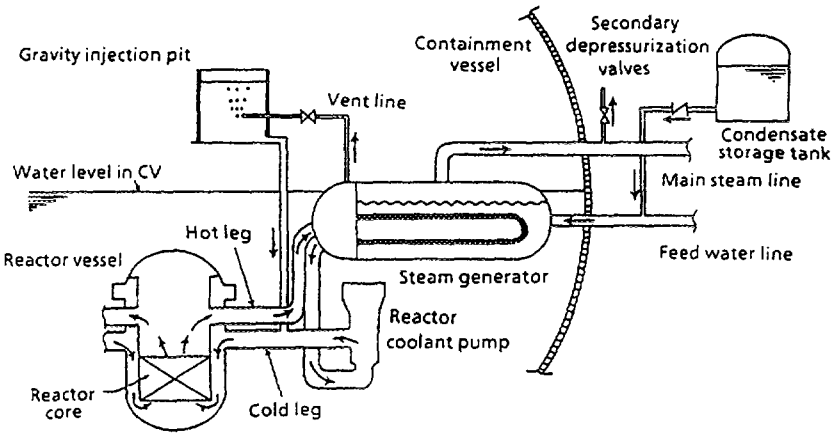


Figure 5.4-4 MS-600 - Passive safety systems

The results of a typical small LOCA analysis are shown in Figure 5.4-5. The primary system pressure decreases from 15 to about 9 MPa in some 700 seconds. Actuation of the PADS brings the pressure down to some 4 MPa and initiation of accumulator injection in another 200 seconds. The accumulator injection inflow amounts to 4-500 kg/s for 75 seconds and continues at about 50 kg/s.

Gravity injection from the gravity injection pit at a flow rate of about 400 kg/s begins at about 1500 seconds. The SADS is actuated at about 900 seconds. As shown, the depressurization of the RCS by the primary and secondary ADS actuates the advanced accumulators and the gravity injection and prevents core uncovery.

Advanced accumulators. The advanced accumulator represents a new accumulator design which includes a fluid flow control device in the lower portion of the tank (Figure 5.4-6). When water injection from the water-filled accumulator is initiated, water will flow out through the main flow standpipe and pass directly through the flow control device to the outlet pipe. When the water level drops below the top of the standpipe, however, the large flow path through the standpipe is broken and the flow switches to the side connection. This path sets up a vortex that causes a high pressure drop and a reduced outflow rate. In this way, an accumulator injection curve with a high initial flow rate followed by a much lower prolonged flow rate has been achieved.

In the latest version of the advanced accumulator, a gas vent system has been connected to the tank. This system can prevent injection of nitrogen gas into the RCS and keeps the RCS pressure below the gravity injection head level. When the water injection has been completed, the gas vent valves are opened and the nitrogen gas is discharged into the containment vessel. This operation will take place before the water level in the tank has dropped below the flow damper.

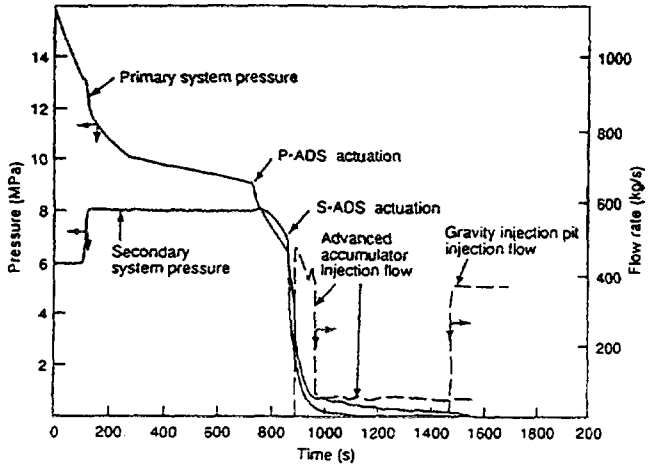


Figure 5.4-5 RCS depressurization and passive injection systems actuation in small LOCA.

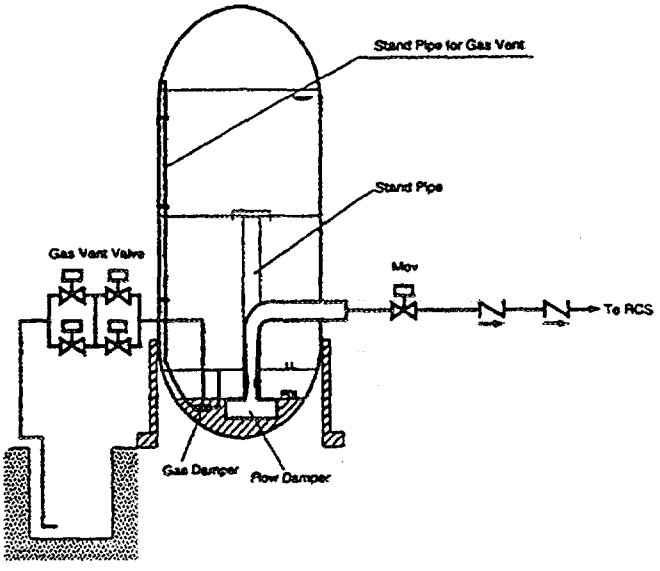


Figure 5.4-6 MS-600 - Advanced accumulator design

Gravity injection pit. The pit which supplies borated water for gravity injection is arranged on the operating floor of the containment vessel. The pit is a compartment in the concrete structure, and it is connected to the RCS cold leg piping. The gravity injection pit is also used for refuelling water storage.

Steam generator heat removal. This system uses the steam generators for cooling of the reactor core. The decay heat from the core (e.g., following a LOCA) is transferred to the steam generators by natural circulation in the primary system, and in order to avoid a possible siphon breaker effect due to accumulation of non-condensable gases in the steam generator tubes a horizontal steam generator design has been chosen.

Non-condensable gases are removed from the loop by means of a vent line on the channel head of the steam generators, instead of letting them into the tubes. The performance of this design feature has been confirmed in a test that simulated accident conditions. Detailed studies of the horizontal steam generator design are continuing.

Active safety systems

The active safety systems consist of charging/safety injection pumps, auxiliary feedwater pumps and their power sources. The operating procedures for them are the same as in currently operating PWR plants, but they have a more limited set of functions and a reduced number of roles to play and so the systems have become simpler.

Some specific design features of the active safety systems are presented below.

Charging/safety injection pumps. The charging/safety injection pumps supply makeup water from the gravity injection pit to the reactor coolant system under high pressure to prevent actuation of the automatic depressurization system (ADS) and subsequent excessive cooling of the RCS in events such as a very small LOCA. Bypass lines are provided in the charging line around the letdown heat exchanger and the charging flow control valve so that it is possible to inject the required flow rate.

Auxiliary feedwater system. The function of the auxiliary feedwater system (AFWS) is to remove heat from the core through the steam generators during any plant condition, when the normal secondary systems are not available. The AFWS consists of two motor-driven and one turbine-driven auxiliary feedwater pumps.

The AFWS is a safety system used to supply water to the steam generators following transients or accidents such as very small LOCAs, loss of the main feedwater supply, rupture of the main feedwater pipe, rupture of the main steam pipe, and rupture of a steam generator tube. When a LOCA accident occurs and

Table 5.4-I. MS-600 - Possible countermeasures for severe accident phenomena

Physical phenomena (threat for containment integrity)	Countermeasure
Over-pressurization (semi-static)	- Reactor coolant system is flooded - Decay heat removal can be done by: Natural circulation through SGs; Containment recirculation coils; Containment external sprays.
Core-concrete interaction	- Cavity flooding line with thermal/ rupture disk
Hydrogen burning/detonation	- Igniter (optional)
Steam explosion	- Labyrinth layout of concrete (prevention of missile impact to containment wall)
Direct containment heating	- Automatic depressurization system (primary and secondary)
Containment bypass	- RHRS is inside containment - ADS prevents release of radioactive inventory from primary to secondary

the reactor coolant system has been depressurized to the level at which the automatic depressurization systems are actuated, the secondary gravity driven feedwater system will be actuated since the secondary side will be depressurized to a pressure level that is too low for the continued operation of the AFWS.

Residual heat removal system. The residual heat removal system (RHRS) of the MS-600 consists of two separate and independent trains which are located inside the containment vessel in order to avoid the possibility of an interface LOCA. Canned-motor pumps are used for the RHRS to make the pumps capable of operating during LOCA situations and to simplify maintenance.

The RHRS is designed to provide the following functions:

- removal of decay heat from the reactor coolant system via the cooling chain consisting of the component cooling water system and the sea water system to the ultimate heat sink, following steam generator cooling during normal operation and in the early cool down phase after a reactor shutdown;
- removal of the decay heat from the RCS via the cooling chain consisting of the component cooling water system and the sea water system to the ultimate heat sink, after 3 days of steam generator cooling following a LOCA.

5.4.6.3 Severe accidents (Beyond design basis accidents)

Severe accident mitigation strategy

Even though the calculated core damage frequency is very low, it has been decided to assume the occurrence of a severe accident and to establish countermeasures for each physical phenomenon which may threaten the containment were established.

The findings are listed in Table 5.4-I; countermeasures are not restricted to only safety-graded systems, but include also the utilization of non-safety class systems.

Severe accident prevention and mitigation

As noted above, the very low calculated core damage frequency shows that the design has a high capability of preventing severe accidents. The prevention features of the safety systems themselves have been discussed in Section 5.4.6.2, and mitigation features are presented in Figure 5.4-7.

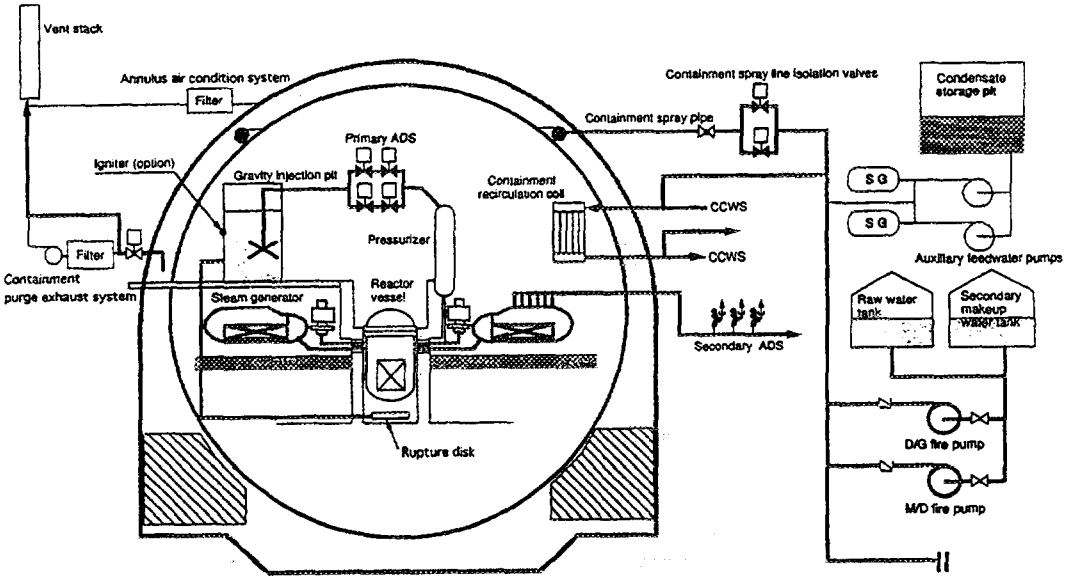


Figure 5.4-7 MS-600 - Severe accident mitigation features

5.4.7 Plant layout

5.4.7.1 Buildings and structures, including plot plan

The general layout of the MS-600 reactor plant is shown on Figures 5.4-8 through 5.4-12. The reactor building and the containment are partly surrounded by an auxiliary building. The access control area, electrical area and other auxiliary system areas are located in the auxiliary building. The main control room and the remote shutdown area are also located in the auxiliary building.

Design requirements

The plant is designed to withstand the impacts of internal and external hazards. With respect to earthquake and explosion pressure waves, the buildings have been strengthened so that the function of safety-grade equipment will not be jeopardized by collapsing structures, and the equipment itself must withstand the dynamic effects inside the buildings.

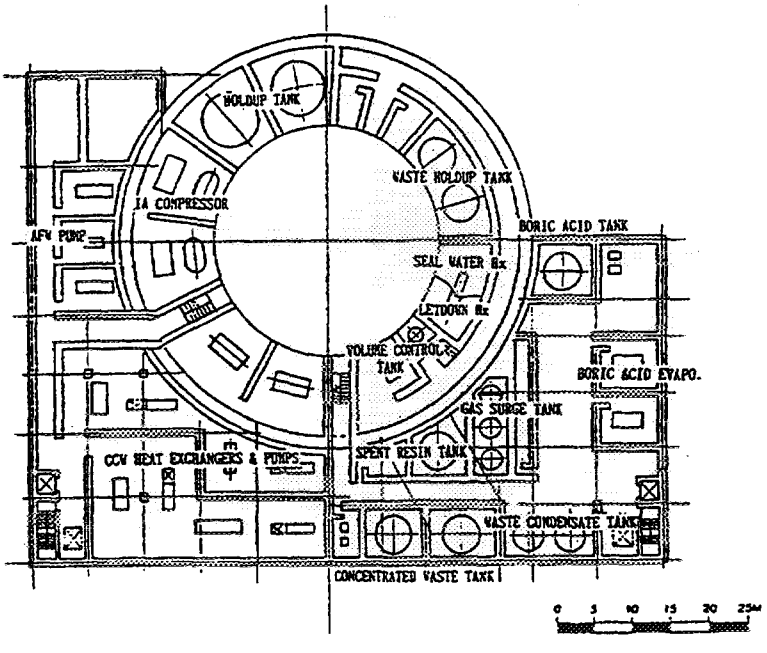


Figure 5.4-8 MS-600 - Building arrangement, 1st floor

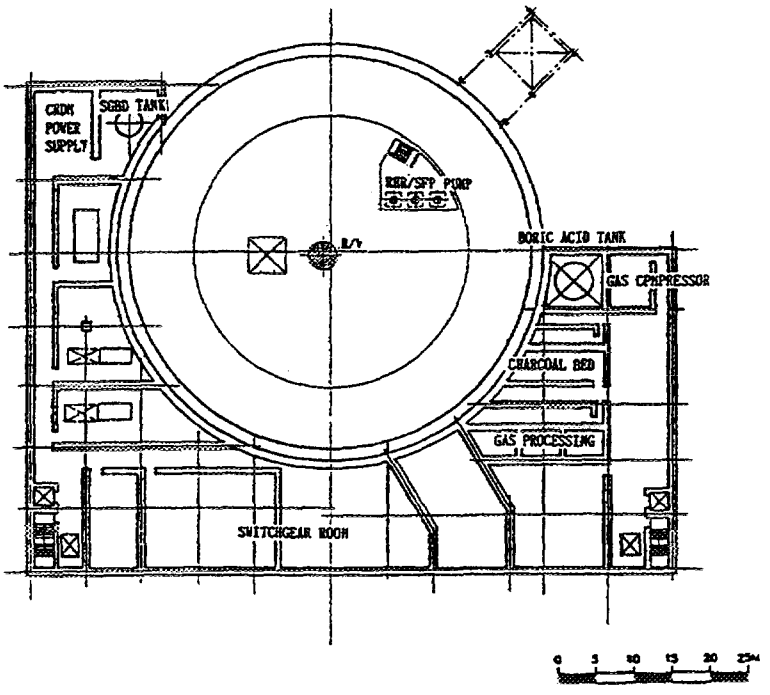


Figure 5.4-9 MS-600 - Building arrangement, 2nd floor

Earthquake. The seismic design is based on the requirements for a Japanese reference site.

Aircraft crash. Design criteria with respect to aircraft crashes are under study.

Internal hazards. Loads from internal events (e.g., fire loads, missile loads, jet impingement loads, and flooding effects) are design-dependent and have not been determined yet. From overall plant layout considerations they should be minimized and simple protection measures should be sufficient toprotect sensitive equipment.

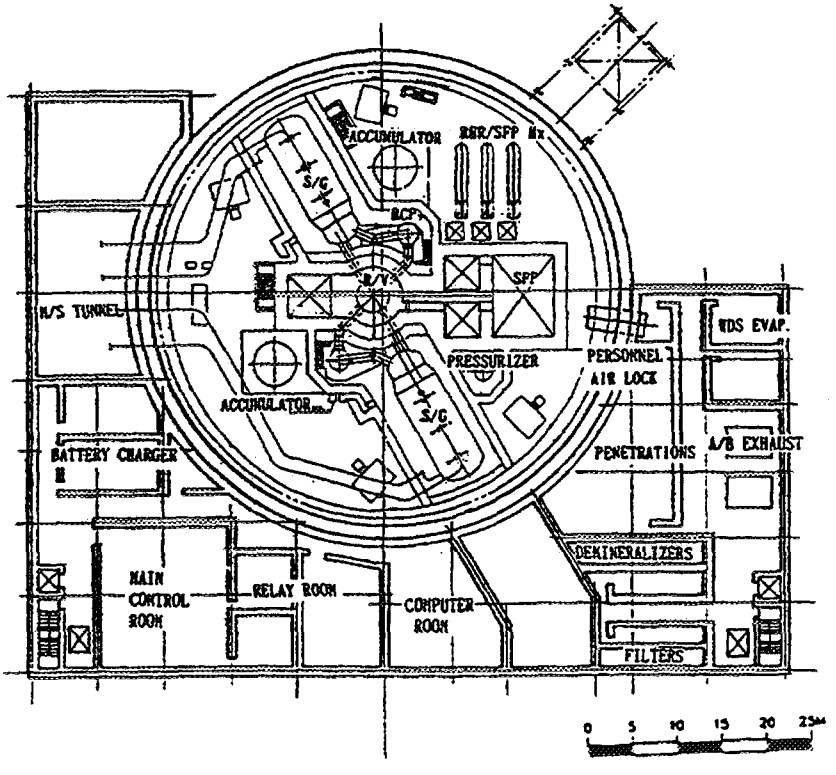


Figure 5.4-10 MS-600 - Building arrangement, 3rd floor

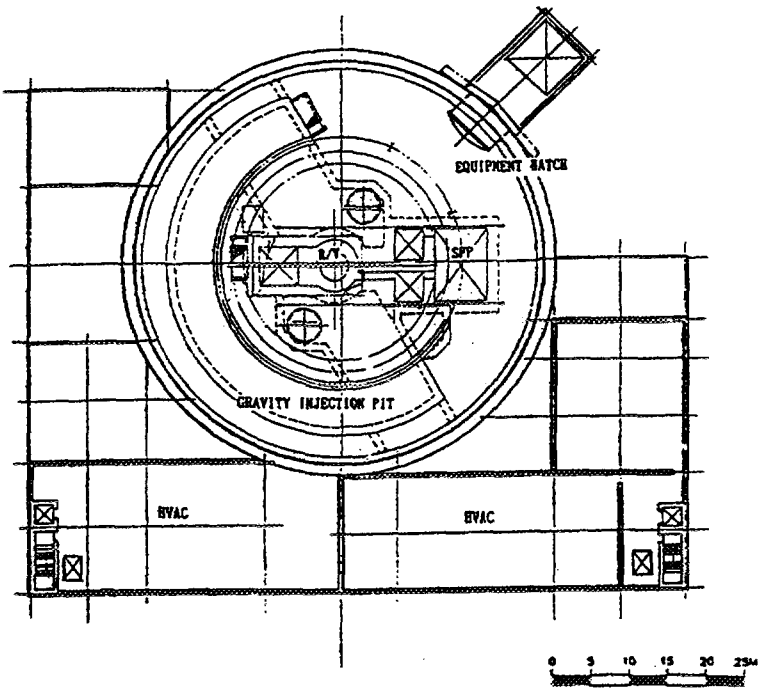


Figure 5.4-11 MS-600 - Building arrangement, 4th floor

Physical separation aspects. Protection against external and internal hazards includes separation of safety-grade systems by divisions and physical protection of the containment enclosing the reactor coolant pressure boundary. By these means, the risk of inadmissible releases or common mode failures of safety-grade systems will be consistent with the deterministic design basis and the probabilistic targets

Radiation protection aspects (accessibility, shielding, ventilation). Besides the requirements concerning severe accident mitigation, the application of radiation protection principles has influenced the plant layout significantly. The MS-600 is designed so that hot (or controlled) and cold (or not controlled) areas are separated.

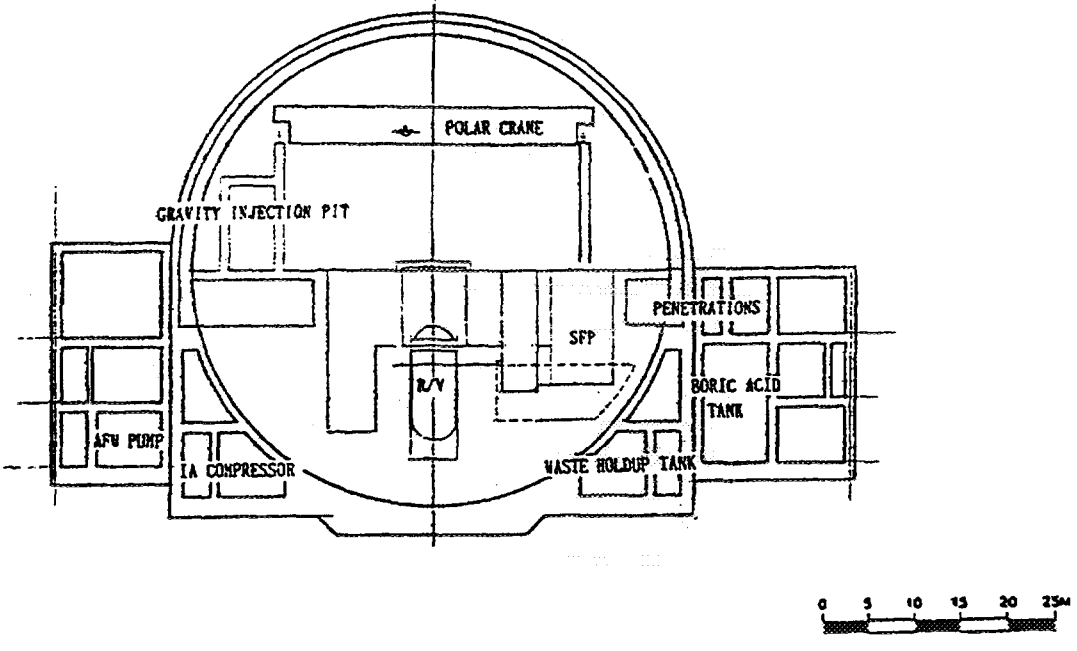


Figure 5.4-12 MS-600 - Building arrangement, vertical section

5.4.7.2 Reactor building

The reactor building is the central building of the plant complex. In essence, it coincides completely with the containment, and thus, the following description of the containment covers also the reactor building.

5.4.7.3 Containment

The MS-600 has a double containment consisting of a steel primary containment and a concrete-filled steel plate secondary containment structure. The primary containment is a 52 meter diameter steel sphere. The spent fuel pit and its cooling system are located inside the containment, together with the gravity injection pit, and the residual heat removal system. The large operating floor area (Figure 5.4-12) is expected to be a big advantage with respect to maintenance activities, and will be an important factor in shortening outage times and reducing radiation exposures for maintenance personnel.

The gravity injection pit which is also used as a refuelling water storage tank, is located on the operating floor, and the loops of the reactor coolant system and the advanced accumulators are located below it. The overhead polar crane is supported from the wall of the gravity injection pit.

The storage pool for spent fuel assemblies is located inside the containment, facilitating fuel transport cask loading and unloading for the transport of spent fuel away from the plant and of new fuel to the plant.

The primary system is arranged symmetrically. Concrete walls are provided between the loops and between the hot and cold legs of each loop to provide protection against consequential failures. The pressurizer is located in a separate compartment. A concrete wall around the entire primary system protects the containment from missiles and reduces the radiation from the primary system to the surroundings.

A water pool for storage of the upper core internals during refuelling, and for the entire core internals during inspection, is provided inside the containment for radiation protection reasons.

The secondary containment, or shield building, is a concrete-filled steel plate structure which can be built quickly and made leaktight. The annulus inside this leaktight structure is vented through a charcoal filter to reduce any radioactive releases to the atmosphere. The annulus vent system consists entirely of passive components.

This type of secondary containment also provides good protection against external missiles.

Containment integrity

The peak value of containment internal pressure in the event of a LOCA is kept below the design pressure by the passive systems alone, and internal spray systems are not provided. Depressurization after 3 days is assumed to be done by means of active components.

Calculation of the containment vessel pressure during a LOCA has been performed to confirm the accident sequence. For the purpose of determining the containment vessel pressure, the mass and energy releases to the containment have also been calculated, assuming a double-ended break of the primary coolant pipe.

The reactor containment vessel is designed so that leak rate tests can be carried out to verify that the total leak rate would not exceed the design value. Also all penetrations, electrical penetrations, air locks, etc., are designed so that leak tests or leak rate tests can be conducted individually or in small groups.

The passive annulus system helps reduce radioactive releases to the environment in the event of accidents. It is designed to remove iodine from any gases that may leak from the primary containment vessel in accident situations.

5.4.7.4 Turbine building

The turbine building houses the main turbine generator and its auxiliaries along with the condensate polishing plant and the condensate/feedwater heating equipment, etc.

The turbine generator is aligned with its longitudinal axis coinciding with the reactor centre line. This arrangement has been adopted to minimize the probability of the reactor building and safety-related equipment being hit by any missiles which may be generated in the unlikely event of a turbine disintegration.

The turbine building foundation is a thick reinforced concrete mat that is supported partly on bed-rock and partly on man-made rock. The parts of the turbine building which are below ground level, including the foundations, are made of reinforced concrete.

A steel frame structure is used for the portions of the building above ground level, designed to withstand all possible loadings, including the crane load.

With respect to the turbine generator arrangement, special emphasis has been placed on a turbine plant arrangement which utilizes space efficiently, not only during plant construction, but also during operation and maintenance of equipment. This has been achieved by providing perimeter and transverse access aisles in the turbine building, strategically located platforms and walkways, and ample space around equipment for disassembly or removal of components as may be necessary during inspection and maintenance.

5.4.7.5 Other buildings

Safeguard buildings

Connections between the safety systems and the reactor coolant system are made as short as possible. The individual trains of the safety systems are arranged radially with respect to the primary loops. Each train of the safety systems is protected against hazards generated in another train by locating each train in a separate area.

The reactor building and the safeguard buildings are protected against external hazards, such as earthquakes. All these buildings are situated on a common raft.

Nuclear auxiliary building

This building mainly houses:

- The gaseous waste system;
- The steam generator blowdown system (including purification equipment);
- The liquid waste system;
- The solid waste system;
- I&C facilities;
- HVAC facilities; and
- Access control facilities.

5.4.8 Technical data

General plant data		
Power plant output, gross	630	MWe
Power plant output, net		MWe
Reactor thermal output	1 820	MWt
Power plant efficiency, net		%
Cooling water temperature	21	°C
Nuclear steam supply system		
Number of coolant loops	2	
Primary circuit volume, including pressuriser		m^3
Steam flow rate at nominal conditions		kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	273/5.8	°C/MPa
Feedwater temperature/pressure	220/-	°C/MPa
Reactor coolant system		
Primary coolant flow rate	9 200	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	291	°C
Coolant outlet temperature, at RPV outlet	325	°C
Mean temperature rise across core	34	°C
Reactor core		
Active core height	3.648	m
Equivalent core diameter	2.9	m
Heat transfer surface in the core		m^2
Fuel inventory	66.7	t U
Average linear heat rate	12.7	kW/m
Average fuel power density	27.3	kW/kg U
Average core power density (volumetric)		kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		

Fuel material	Sintered UO_2	
Fuel assembly total length	4 200	mm
Rod array	square, 17×17	
Number of fuel assemblies	145	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	24	
Number of spacers	9	
Enrichment (range) of first core		Wt%
Enrichment of reload fuel at equilibrium core		Wt%
Operating cycle length (fuel cycle length)	20	EFPM
Average discharge burnup of fuel	50 000	MWd/t
Cladding tube material	Zr Alloy	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly	680	kg
Active length of fuel rods	3 648	mm
Burnable absorber, strategy/material	NA	
Number of control rods	45	
Absorber rods per control assembly	24	
Absorber material	NA	
Drive mechanism		
Positioning rate		mm/min
Soluble neutron absorber	boron	
Reactor pressure vessel		
Cylindrical shell inner diameter	4 400	mm
Wall thickness of cylindrical shell	200 + 6	mm
Total height	11 800	mm
Base material: cylindrical shell	Low alloy steel forging	
RPV head	Low alloy plate	
liner	Stainless steel	
Design pressure/temperature	17.2/343	MPa/°C
Transport weight (lower part)		t
RPV head		t

Steam generators

Type	Horizontal, U-tube heat exchanger	
Number	2	
Heat transfer surface	4 850	m^2
Number of heat exchanger tubes	4 206	
Tube dimensions	19.05	mm
Maximum outer diameter	5 100	mm
Total height	16 000	mm
Transport weight	425	t
Shell and tube sheet material	SA533/SA508	
Tube material	Inconel 690	

Reactor coolant pump

Type	Single-stage, centrifugal pump	
Number	2	
Design pressure/temperature	17.5/343	MPa/°C
Design flow rate (at operating conditions)	4 600	kg/s
Pump head		
Power demand at coupling, cold/hot		kW
Pump casing material	Ferritic steel with cladding	
Pump speed	1 190	rpm

Pressuriser

Total volume	28.3	m^3
Steam volume: full power/zero power	11.3/21.2	m^3
Design pressure/temperature	17.6/360	MPa/°C
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter	2 356	mm
Total height	9 411	mm
Material	16MND5/20MnMoNi55	
Transport weight	62	t

Pressuriser relief tank

NA

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Dry, double wall	
Overall form (spherical/cyl.)	spherical	
Dimensions (diameter/height)	52/52	m
Free volume	48 000	m^3
Design pressure/temperature (DBEs)	~ 0.5/	MPa/°C
(severe accident situations)	~ /	MPa/°C
Design leakage rate	< 1	vol%/day
Is secondary containment provided?	yes, space between the walls	

Reactor auxiliary systems

Reactor water cleanup,	capacity	8	kg/s
	filter type		
Residual heat removal,	at high pressure	120	kg/s
	at low pressure	60	kg/s
Coolant injection,	at high pressure	8	kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage	500	kV
	rated capacity		MVA
Plant transformers,	rated voltage	500	kV
	rated capacity		MVA
Start-up transformer	rated voltage	77	kV
	rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)		6.6	kV
Number of low voltage busbar systems		8	
Standby diesel generating units: number		1 non-safety/ 2 safety	
	rated power	4.8/ 0.6	MW
Number of diesel-backed busbar systems		2 / 2	
Voltage level of these		6.6 / 0.44	kV ac
Number of DC distributions		6	
Voltage level of these		125	V dc
Number of battery-backed busbar systems			
Voltage level of these			V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	3 cylinder, 4 flow, exhaust type	
Number of turbine sections per unit (e.g. HP/LP/LP)	HP/ LP/ LP	
Turbine speed	1 500	rpm
Overall length of turbine unit	55	m
Overall width of turbine unit	14.5	m
HP inlet pressure/temperature	5.5/270.1	MPa/°C

Generator

Type	3-phase, turbogenerator	
Rated power	650	MVA
Active power	585	MW
Voltage	19	kV
Frequency	50	Hz
Total generator mass	524	t
Overall length of generator	18	m

Condenser

Type	Rectangular, single flow	
Number of tubes	40 500	
Heat transfer area	48 900	m^2
Cooling water flow rate	38	m^3/s
Cooling water temperature	21	°C
Condenser pressure	402	hPa

Condensate pumps

Number	3	
Flow rate	305	kg/s
Pump head	0.7	MPa
Temperature	34.3	°C
Pump speed	1 000	rpm

Condensate clean-up system

Full flow/part flow	Full flow
Filter type	mixed bed

Feedwater tank

Volume	300	m^3
Pressure/temperature	1.1/183.2	MPa/ °C

Feedwater pumps

Number	3	
Flow rate	505	kg/s
Pump head	5.2	MPa
Feedwater temperature	183.2	°C
Pump speed	4 900	rpm

Condensate and feedwater heaters

Number of heating stages	6
Redundancies	0

5.4.9 Project status and planned schedule

The conceptual design work and the basic design work have already been completed, and detailed design is now in progress. A test programme is underway to confirm important design features.

References

ANP '92

ICONE-2, Volume 1

5.5 SBWR, GE

5.5.1 Introduction

The Simplified Boiling Water Reactor (SBWR) represents a complete BWR plant design with a nominal output of 600 MWe that has been under development by General Electric since the mid-1980s, as one of the two mid-size designs of the US ALWR Development Program.

The design is based on a natural circulation BWR, with a nominal core power output of 2000 MWth, and incorporating a number of innovative featuresto achieve plant simplifications. Among others, it includes introduction of passive safety features, instead of or as supplements to the traditional active safety systems, to ensure the basic safety in accident situations.

A standard preliminary safety analysis report for the design was submitted to the US NRC for review, and work proceeded with preparation of supplementary topical reports on licensing issues. The schedules aimed at approval of all licensing topical reports in 1996, issuance of a final design approval (FDA) by the NRC in mid-1998, and a design certification in late 1999.

The basic design work was well underway, and substantial R&D work and verification testing were performed on an international level. The R&D efforts were focused on analysis and tests considered necessary for certification, and directed toward demonstration of the passive safety systems, both individually and in a simulated SBWR integrated systems environment, and data acquisition, as well as validation of the TRACG code analysis with respect to modelling of SBWR transient phenomena and containment performance.

The integral response of the SBWR during the post-blowdown and Gravity Driven Cooling System (GDCS) operation phase was demonstrated in the Gravity Driven Cooling System Integral Systems Test (GIST) on a 1:500 scale, with a focus on the important RPV response. The GIST programme also provides an extensive database for qualifying the TRACG code for a wide range of SBWR condition and parameters.

The Passive Containment Cooling System (PCCS) performance under a range of SBWR conditions following a LOCA has been proven by tests in GIRAFFE, a 1:400 scale facility at Toshiba, Japan, representing major SBWR compartments in the nitrogen-inerted containment. In addition, single tube condensation test programmes were conducted at UC, Berkeley. The thermal, hydraulic and mechanical performance of the PCCS will be demonstrated at full-scale in the PANTHERS facility. Extensive instrumentation has been added to facilitate qualification of the TRACG code and also monitor individual tube performance.

Further tests on the integral response of the SBWR were planned to be performed in PANDA, a 1:25 scale, full height facility which simulates reactor vessel, wetwell, drywell, GDCS pool, IC and PCCS pool and heat exchangers.

The feasibility of realizing larger size units using the same basic concepts was also investigated, e.g., in Japan, and in mid-1996 General Electric announced that further development work on the SBWR had been abandoned, including the licensing review activities. General Electric had decided to independently pursue a similar design with a larger output.

The description of this new design version is included in Chapter 4, as 4.5 ESBWR. The design objectives and features of the ESBWR are largely identical to those of the SBWR. Therefore, a complete description of the SBWR, beyond the highlights presented in Section 5.5.2, is not included in this report; this note merely serves as a reminder and a reference.

5.5.2 Some highlights of the SBWR design

The design of the SBWR addresses a complete power plant design with a nominal power output of 600 MWe. The SBWR is designed to operate with natural circulation of the primary coolant and incorporates innovative concepts to further simplify an inherently simple direct cycle nuclear plant.

The core configuration consists of 732 bundles - 648 interior bundles and 84 peripheral bundles. The inlet orifice of the peripheral bundles is restricted in order to preferentially force flow through interior, high power bundles. The rated core power is 2000 MWt, which corresponds to a 41.5 kW/l power density. The lower power density results in improved fuel cycle costs and greater manoeuvrability. The SBWR is a natural circulation reactor, and the reactivity control is maintained by movement of control rods and by burnable poison in the fuel.

The reactor pressure vessel (RPV) is a vertical, cylindrical pressure vessel, with a removable top head, and head flanges, seals and bolting, and with venturi-shaped flow restrictors in the steam outlet nozzles.

The RPV is 6 m in diameter, with a wall thickness of about 158 mm with cladding, and 24.5 m tall from the inside of the bottom head (elevation zero) to the inside of the top head. The bottom of the active fuel location is 3750 mm from elevation zero and the active core is 2743 mm high. The RPV height permits natural circulation driving forces to produce abundant core coolant flow by increasing the internal flow-path length relative to earlier BWRs by means of a long "chimney" in the space which extends from the top of the core to the entrance to the steam separator assembly. The large RPV volume provides a large reserve of water above the core, which translates directly into a much longer period of time (compared to earlier BWRs) before core uncovery can occur as a result of loss of feedwater flow or a LOCA. This gives an extended period of time during which automatic systems or plant operators can re-establish reactor inventory control using any of several normal, non-safety-related systems capable of injecting water into the reactor. Timely initiation of these systems precludes the need for activation of emergency safety equipment. The large RPV volume also reduces the reactor pressurization rates that develop when the reactor is suddenly isolated from the normal heat sink which eventually leads to actuation of the safety-relief valves.

Important new features introduced in the design are the isolation condenser system (ICS), the gravity-driven cooling system (GDCS), and the passive containment cooling system (PCCS).

The ICS removes decay heat after any reactor isolation during power operations. Decay heat removal limits further pressure rises and keeps the RPV pressure below the set point of the pressure relief system. The ICS consists of three independent loops, each containing a heat exchanger that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC/PCC pool which is vented to the atmosphere.

The GDCS provides, in conjunction with the automatic depressurization system (ADS), emergency core cooling in case of a LOCA. When a low level signal is received, the ADS will depressurize the reactor vessel and the GDCS will inject sufficient cooling water to maintain the fuel cladding temperatures below temperature limits defined in 10CFR 50.46. In the event of a severe accident that results in a core melt with the molten core in the lower drywell region, GDCS will flood the lower drywell cavity region with the water inventory of the three GDCS pools and the suppression pool (SP).

The PCCS maintains the containment within its pressure limits for design basis accidents such as a LOCA. The system is passive with no components that move. The PCCS consists of three low pressure totally independent loops, each containing a steam condenser (passive containment cooling condenser that condenses steam on tube side and transfers heat to water in a large cooling pool (IC/PCCS pool), which is vented to atmosphere.

5.6 WWER-640 (V-407), ATOMENERGOPROJECT/GIDROPRESS, RUSSIAN FEDERATION

5.6.1 Introduction

Ensuring the safety of the personnel, the population and the environment against radiation effects is used as the basis for the design. The prescribed doses of exposure, and the standards for the release of radioactive substances and their content in the environment, should not be exceeded under normal operation, anticipated operational occurrences, and in design and beyond-design-basis accidents during the 50-60 years' service life of the plant.

Operating limits for fuel cladding damage are as follows:

- up to 0.1% of fuel rods with flaws of gas leaktightness, and up to 0.01% of fuel rods with direct contact between fuel and coolant,
- 7.4E10 Bq/m^3 primary coolant iodine nuclide radioactivity,
- the iodine nuclide radioactivity in secondary side water of any steam generator should not exceed 1.5E4 Bq/m^3 under normal conditions and operational occurrences. The estimated probability of the operating limits being exceeded is less than 1E-2 per reactor-year.

Fuel element damage leading to:

- a considerable amount of radioactivity release from fuel rods,
- a considerable steam-zirconium reaction progression (considerable from the standpoint of fulfillment of the safety insurance requirements mentioned above),
- a fuel material escape out of the cladding preventing core cooling and post-accident removal,

are design limits that should not be exceeded in design basis accidents. The estimated probability of the design limits being exceeded must be less than 10^{-5} per reactor-year. The estimated probability of considerable fuel damage leading to the necessity of an evacuation of the population from the area prescribed by the relevant guides is specified to be less than 10^{-7} per reactor-year.

The safety of the nuclear power plant will be ensured by a consistent implementation of the "defence-in-depth" principle based on the application of a system of barriers on the path of spreading ionizing radiation and radioactive substances into the environment, as well as of a system of engineered safeguards and of organizational provisions for the protection of these barriers. The consistent implementation of the "defence-in-depth" principle implies:

- installation of successive physical barriers on the path of spreading the radioactive substances: fuel matrix, fuel element cladding, primary circuit boundary, containment;
- taking into account all postulated initial events that can lead to a loss of efficiency of these barriers;
- determination, for each postulated event, of design measures and actions of operating personnel required to keep the integrity of the barriers mentioned, and to mitigate the consequences of a damage of such barriers;
- minimization of the probability of accidents resulting in an escape of radioactive substances; and
- redundancy and diversity of safety systems, and physical separation of safety system trains.

The nuclear power plant considered is of the evolutionary type. The principal technical decisions have been supported by operating experience for more than 300 reactor-years with nuclear power plant of the WWER-440 type, including the Loviisa and Paks plants in Finland and Hungary which are known to be among the best performing nuclear power plants in the world measured by their load factors. Operation of WWER-1000 plants for more than 90 reactor-years also contributes to the base of experience.

The new design features are envisaged to be verified experimentally at a large-scale test facility (1:27 volume and power scale). The design is developed in accordance with the latest Russian safety regulations for nuclear power plants (references [1] and [2]) which meet modern world requirements. The design

organizations involved are: OKB "Gidropress", Russian National Research Centre "Kurchatov Institute" and LIAEP, the well-known designers of WWER nuclear power plants. QA requirements of the IAEA and the international standard ISO 9000 are taken into account in the design.

In the plant safety concept, the world's modern trends in nuclear power plant safety improvements are considered in order to meet, for as long a period as possible, the current and future requirements for nuclear power plant safety which are constantly becoming more strict. The design passed the international examination at the nuclear power plant design competition in St. Petersburg in 1993.

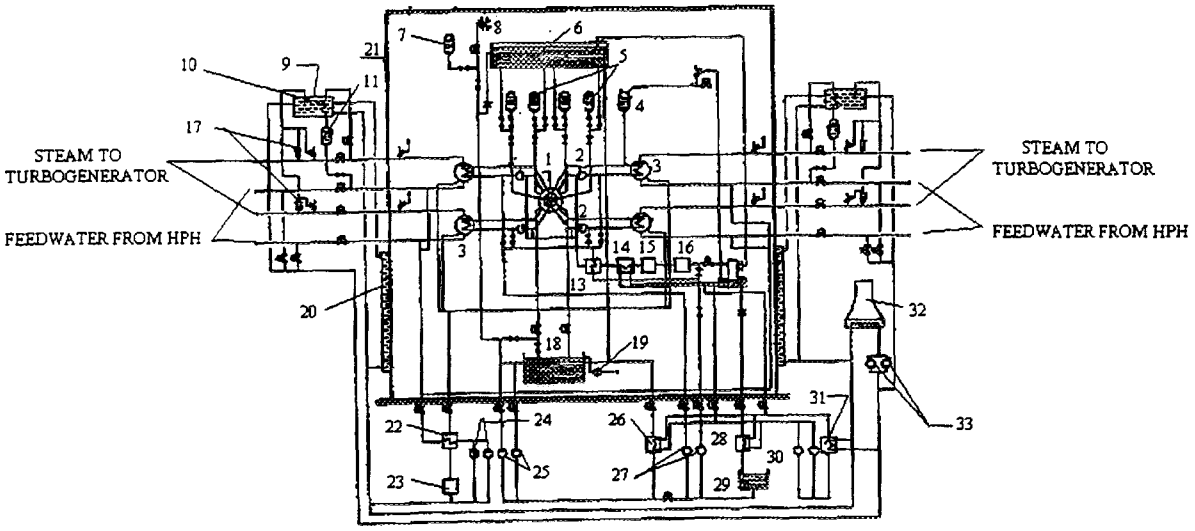
The principal features characterizing the safety philosophy accepted in the design are as follows (absolute figures will be presented in other sections):

- considerable decrease of specific fuel power (it is 1.25 times less than in the Loviisa nuclear power plant reactor and 1.5 times less than in a standard WWER-1000 reactor) due to an increased number of fuel assemblies,
- the fluence to the reactor vessel considered is one order of magnitude less over 60 years than that to the vessel of the standard WWER-1000 reactor over 40 years,
- the possibility of providing subcriticality with solid control rods at any moment of the life-time for a coolant temperature decrease down to 100 °C and assuming complete replacement of the boric acid in the primary circuit with pure condensate,
- retaining the large ratio of the primary and secondary coolant volumes to the reactor power typical for WWER-440 reactors (1.5-2 times more in comparison with the standard WWER-1000 reactor and Western PWRs, which provides softer temperature conditions for the core and the whole nuclear power plant equipment under transient and accident conditions),
- simplification of the operating and layout features for safety systems and all other systems of the plant (in comparison with a WWER-440 the number of pumps and compressors is reduced 4 times, the number of shut-off valves 3 times; there is a 2 time reduction in the number of high-and-low pressure tanks, and the number of sealed process penetrations is reduced by a factor of 4),
- application of horizontal steam generators with large water inventories and better conditions for natural circulation in the primary circuit in comparison with vertical steam generators,
- application of an emergency core cooling system based on the principle of passive operation that provides for the possibility of long-term residual heat removal after LOCA accidents taking also into account a possible station blackout,
- application of passive systems for residual heat removal from the reactor plant in case of a station blackout (transient),
- application of passive systems of residual heat removal from the containment,
- provision of a large water inventory inside the containment (about 2000 m^3) required to form the emergency heat removal pool, the water level of which rises above the hot legs after flooding,
- application of an inner, sealed steel shell, enclosed by an outer concrete protective shell, and both together constituting the containment system,
- application of diagnosis systems for equipment and systems important to safety for on-line diagnosis during operation and for periodic inspections after shutdown,
- application of an automatic control system of improved reliability, with self-diagnosis, and an expert system for giving advice to the operator,
- redundancy, diversity, physical separation of safety systems as part of defence in depth.

5.6.2 Description of the nuclear systems

5.6.2.1 Primary circuit and its main characteristics

The design of the primary circuit uses a 4-loop configuration with horizontal steam generators. The design of the steam generators with stainless steel collectors is similar to that of WWER-440. The flow diagram and a drawing of the reactor plant are shown in Figures 5.6-1 and 5.6-2. Name, list and quantity of the reactor components shown in Figure 5.6-1 are given in the legend.



LEGEND:

No	Name	Quantity	No	Name	Quantity
1	Reactor	1	18	Fuel pond	1
2	Primary coolant pump	4	19	Screen filter	2
3	Steam generator	4	20	Containment cooler	2
4	Pressurizer	1	21	Containment	1
5	HP ECCS hydrotank	4	22	Regenerative heat exchanger	2
6	ECCS atmospheric tank	4	23	Water treatment plant	1
7	Tank for Iodine fixing system	2	24	Pump of the secondary makeup-blowdown system	4
8	Sprinkler	1	25	Fuel cooling system pump	4
9	Demineralized water storage tank	4	26	Fuel cooling system heat exchanger	2
10	Heat exchanger	4	27	Primary makeup-blowdown system pump	4
11	Heat Removal System Buffer		28	Heat exchanger	1
12	Quick-acting pressure reducing device	4	29	Water tank in auxiliary system	1
13	Regenerative heat exchanger	4	30	Intermediate cooling circuit pump	4
14	Aftercooler	4	31	Intermediate cooling circuit heat exchanger	2
15	Coolant treatment plant	1	32	Dry cooling tower	1
16	Electrolyzer	2	33	Pump	4
17	Bubbler-degasser	1			

Figure 5.6-1 Flow diagram of the reactor system

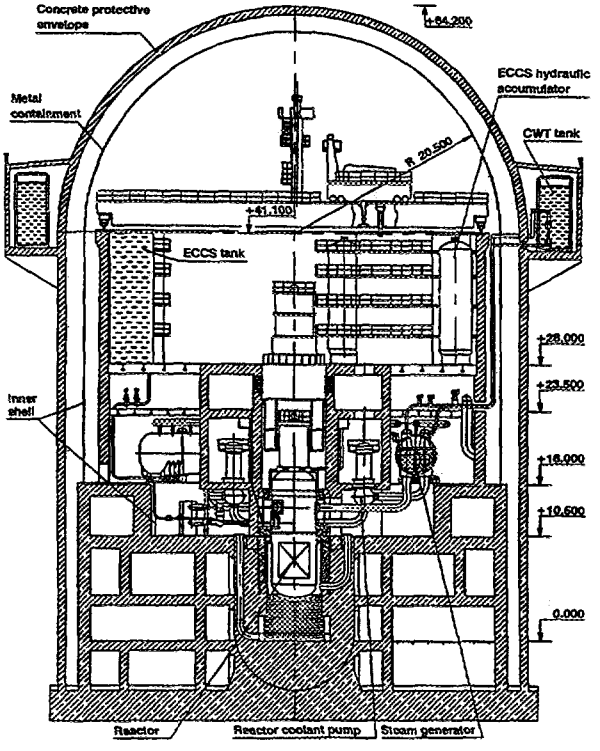


Figure 5.6-2 Reactor building

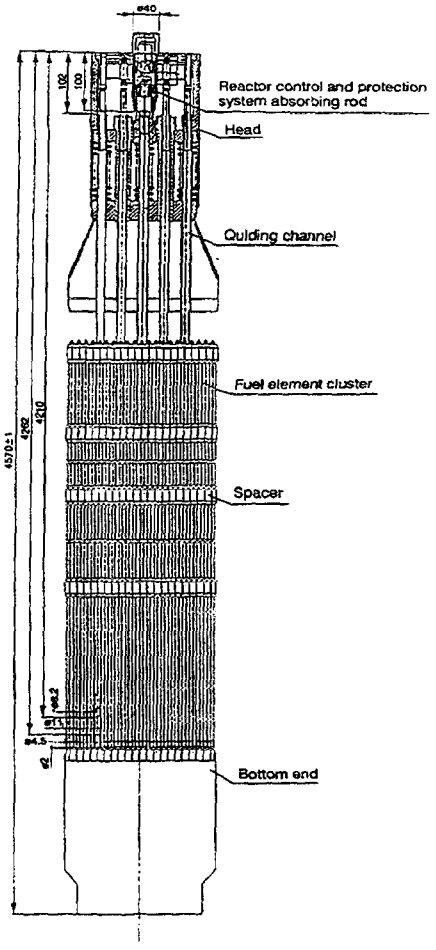


Figure 5.6-3 Fuel assembly

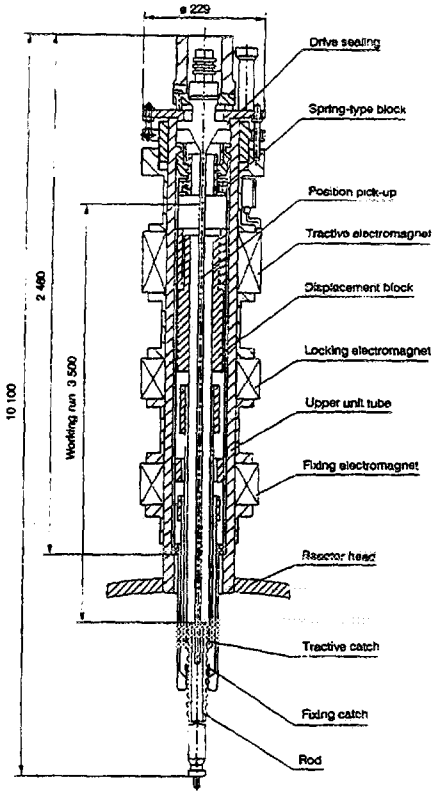


Figure 5.6-4 Control rod drive

5.6.2.2 Reactor core and fuel design

The core comprises 163 fuel assemblies. A drawing of the fuel assembly is shown in Figure 5.6-3. In the reactor scram system 121 control rods are used. A step-by-step electromagnetic (magnetic jack) drive with position indicator is used to move the control rods. The control rod assemblies comprise 18 absorber rods suspended on a brace. The absorber rods are made of stainless steel tubes filled with an absorber material which varies along the length, boron carbide is used for the upper part and dysprosium titanate for the lower. The rod cladding is sealed with end caps. The control rod drives are installed on the reactor top head. A drawing of the control rod drive is shown in Figure 5.6-4. Effective operation time between refuellings is 293 full power days. Average burnup of the fuel unloaded is 40 MWdays/kgU. The number of fresh assemblies loaded during annual refuelling is 36.

5.6.2.3 Fuel handling and transfer systems

The fuel handling and transfer systems are intended for loading of fresh fuel assemblies and replacing spent fuel assemblies, and shutdown absorber rods.

The fuel handling and transfer systems include the following compartments with necessary equipment: the reactor concrete well; the fuel pool; and the transfer compartment.

The fuel pool is located in the vicinity of the reactor concrete well; they are connected by a transfer channel designed for transportation of one fuel assembly at a time. The fuel pool is provided with storage structures for spent fuel assemblies; these structures consist of separate sections designed for storing of fuel assemblies and sealed containers for failed fuel assemblies.

The transfer compartment is provided with an universal nest for location of a plant-internal transportation unit for fresh fuel assemblies and for a transport cask for spent and decayed fuel assemblies. The transfer compartment is connected to the fuel pool by the transfer channel through which loading of fresh fuel assemblies take place. The fresh fuel assemblies are taken out of the plant-internal transportation unit by the refuelling machine and installed in the core in accordanse with the core refuelling chart.

After unloading from the core, the spent fuel assembly is placed in a container with a fuel assembly defect monitoring system, and then transported to the storage structure or the sealed container for spent and decayed fuel assemblies in the fuel pool depending on the results of the defect inspection.

Manipulations for control rod assembly replacements are similar to those for fuel handling; control rod clusters and burnable poison rods extracted from the reactor core are installed in empty fuel assemblies or in storage structures in the fuel pool for storage.

The refuelling machine handles only one fuel assembly, one control rod cluster, or one fuel assembly with the control rod cluster inside it, at a time. Heat removal from the reloaded fuel assemblies is accomplished by the fuel pool cooling system.

5.6.2.4 Primary components

Reactor pressure vessel

A schematic drawing of the reactor assembly is shown in Figure 5.6-5 and Figure 5.6-6 a cross-section through the reactor vessel. The reactor vessel of the earthquake-proof WWER-1000 (V-392) reactor is adopted for the V-407 reactor. The vessel design lifetime is 60 years; for the rest of the reactor equipment, except equipment that is exchangeable during the service life (in accordance with its prescribed lifetime), the design lifetime is 50 years.

The reactor assembly consists of the following parts: the reactor vessel proper with main closure head and associated details; an upper unit with 121 control rod drive mechanisms; reactor internals (core support structures, outer core structure, upper protective tube structure); and the reactor core.

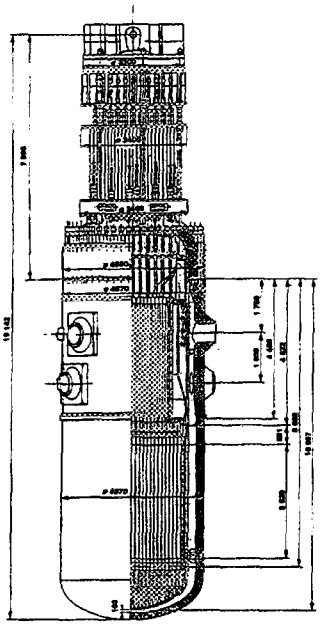


Figure 5.6-5 Reactor vessel assembly

The reactor vessel consists of a flange, upper and lower nozzle zone shells, support shell, cylindrical shell and elliptic bottom, welded together by circumferential seams. The vessel has two rows of nozzles of 620 mm inner diameter; four nozzles in each row. In addition, there are four nozzles of 170 mm inner diameter for the pipeline connections to the ECCS hydroaccumulators in the upper row and four nozzles of 125 mm inner diameter for the pipeline connections to the ECCS tanks in the lower row; the upper row also includes an instrumentation nozzle.

The inner surfaces of the vessel and the nozzles are plated with a corrosion-resistant layer. A separation ring is welded to the vessel inside between the upper and lower nozzle rows for separation between the inlet and outlet coolant flows, and a support rib is made on the vessel support shell for vessel attachment on the support structure.

The reactor vessel is installed in the concrete well on the support framework; the support rib of the vessel is installed and fixed on the support framework ring.

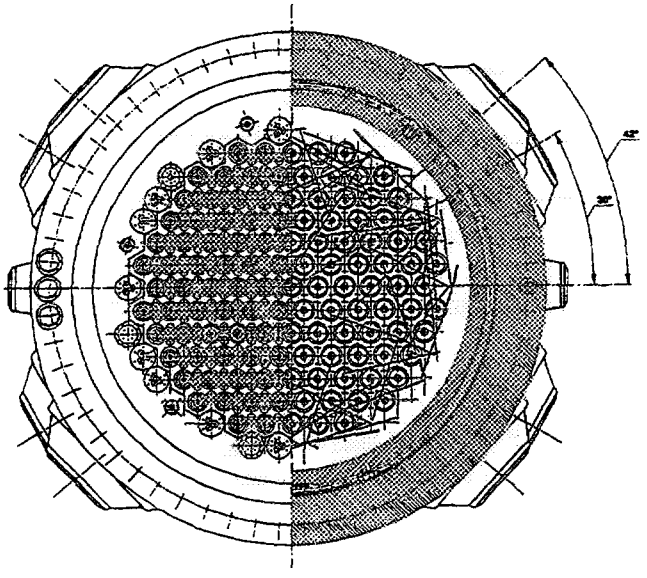


Figure 5.6-6 Reactor vessel cross-section

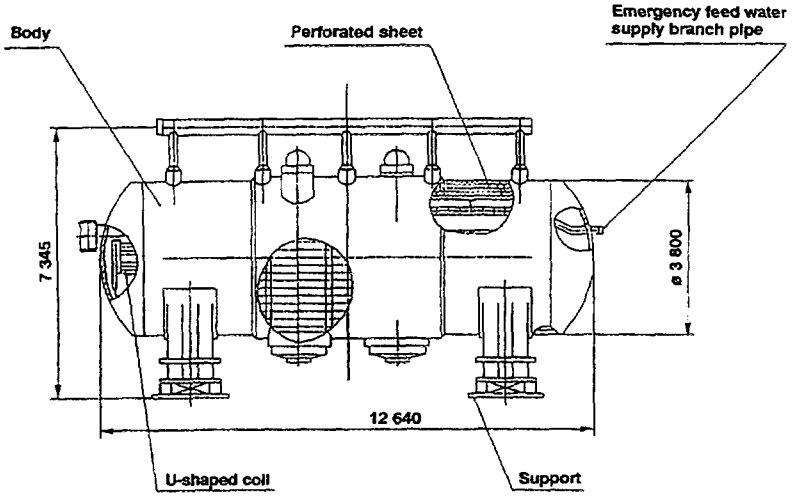


Figure 5.6-7 Steam generator

Reactor internals

Structures providing support and positioning of the core and the control rods are installed inside the vessel; the internal structures including the core support structures, outer core structure, and upper protective tube structure also serve as coolant flow guides for the core heat removal.

Steam generators

The steam generator is of the horizontal type (Figure 5.6-7). For inspection of internals, hatches of 500 mm diameter on both elliptic bottoms as well as hatches of 1000 mm diameter in the cylindrical part of the steam generator are provided

Pressurizer

The pressurizer is the same as for the WWER-1000 design. The pressurizer with primary pressurization system is designed for maintaining primary system pressure within acceptable limits in all reactor plant design conditions. The pressurizer proper is a vertical vessel mounted on a cylindrical support structure. There are nozzles for spray pipelines and safety valve steam discharge line on the upper head, and a nozzle for connection between the pressurizer and the hot leg is provided in the vessel bottom.

Nozzles for level gauges and casings for thermo-resistors measuring coolant temperature inside the vessel are located on its cylindrical shell.

A spray device and a tubular electric heater are located inside the pressurizeer. The spray device is intended for water spraying in the steam volume and condensation of steam; it is made as a discharging header fastened to the top of the vessel. The inside surface of the pressurizer is plated with corrosion resistant stainless steel. All pressurizer internals are made of austenitic stainless steel.

The primary system overpressure protection system is designed to protect reactor plant equipment and pipelines against inadmissibly high coolant pressure in accident situations. The system can also be used for forced depressurization of the primary system. It consists of: two safety valves, a relief valve and pipelines connecting them with the pressurizer; and a bubbler-degasifier and pipelines connecting it with the safety and relief valves.

Reactor coolant pumps

The reactor coolant pump (RCP) is of the centrifugal type in a spherical case (Figure 5.6-8. Lubrication and cooling of the RCP are performed with water. A non-combustible lubricant is used in the electrical motor.

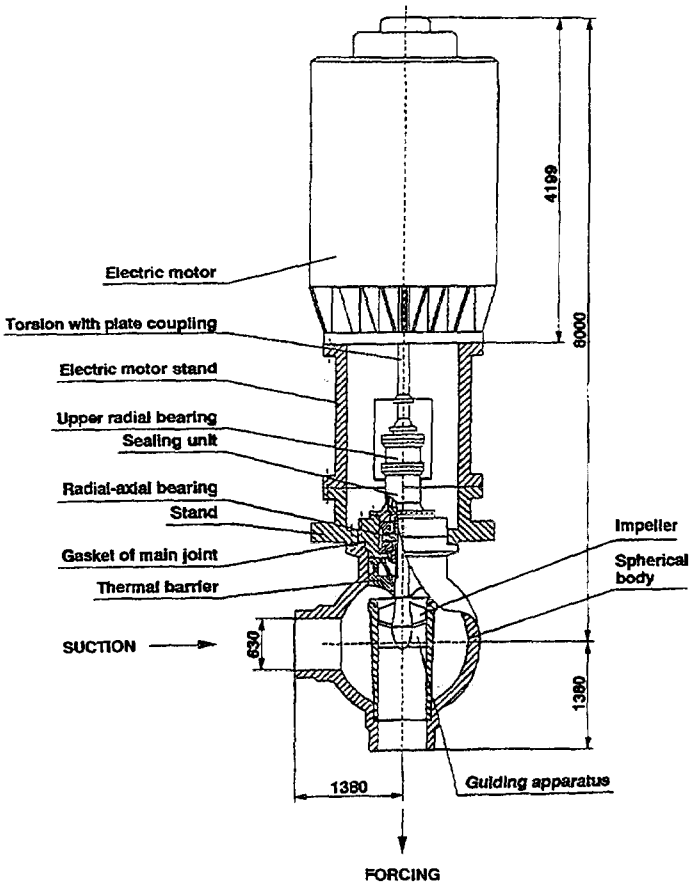


Figure 5.6-8 Primary coolant pump

Main coolant lines

The main coolant lines consist of four loops for coolant circulation. The inner diameter of the main pipelines is 620 mm. The loops do not have loop seals and consist of straight tubes and steep bends on sections of connections of steam generators and main coolant pumps.

The primary equipment layout and the passive heat removal system provide core residual heat removal via the steam generators to the demineralized water storage tank and further to the atmosphere by natural circulation.

5.6.2.5 Reactor auxiliary systems

Chemical and volume control system

No information provided.

5.6.2.6 Operating characteristics

The nuclear power plant unit is designed taking into account the grid needs imposing a high capability to follow grid loads in conditions with rather rapid and deep plant load changes.

5.6.3 Description of turbine generator plant system

5.6.3.1 Turbine generator plant

No information provided.

5.6.3.2 Condensate and feedwater systems

No information provided.

5.6.3.3 Auxiliary systems

No information provided.

5.6.4 Instrumentation and control systems

5.6.4.1 Design concepts including control room

The main control room is the main point point for monitoring the main processes of the nuclear power plant unit and for plant staff controlling it, in accordance with assigned operation tasks. The instrumentation and control design concept provides work positions for operators of the reactor, the turbine and auxiliary equipment, and for a shift supervisor responsible for coordinating the plant unit activities during all operation conditions.

The work positions for control of special engineering means ensuring plant safety, including auxiliary power supply systems, are separated from the normal work positions. A reserve control room for the plant unit is provided from which it is possible to ensure a reliable reactor shutdown to cold and subcritical state and maintain it in this state as long as necessary, including means for initiation of safety systems and devices for receiving information on reactor conditions.

Plant process control systems fulfill the automatic control of the following main controlled parameters:

- neutron flux in the core
- primary pressure
- secondary pressure
- water level in the steam generators
- water level in the pressurizer

The design value of the reactor neutron flux is maintained within $\pm 2\%$ of its nominal value by means of the control bank of neutron absorbers, consisting of several rod cluster control assemblies.

The design value of the primary pressure is maintained within ± 3 bar (0.3 MPa) by the pressurizer electric heaters and by the valves on the coolant injection line from the reactor coolant pump (RCP) exit side to the steam volume of the pressurizer.

The design value of the secondary pressure is maintained within ± 2 bar (0.2 MPa) by keeping the appropriate balance between reactor power and steam flow from the steam generators to the turbine or to the steam dumping devices.

The design value of the water level in the steam generators is maintained within ± 50mm of its nominal level by means of the steam generator feedwater supply controller, actuating the control valve on the steam generator feedwater line.

The design value of the water level in the pressurizer is maintained within ± 150 mm by the level controller, actuating the control valves located on the make-up line.

Protection against transients due to the introduction of reactivity is secured by the operation of the emergency protection (shutdown) system in response to a signal of reaching the neutron flux setting or in response to a signal of reaching the setting of reactor period decrease. Within the operating ranges the reactor's neutron power and the process parameters are maintained automatically by the reactor power controller (ARM-system). Any changes in the process variables outside the control band are made by the operators in deviation of the parameters:

- up to the set-points (PZ-2) the upward motion of the control rods is stopped;

- up to the set-points (PZ-1) type 2 preventive protection is actuated; the power reduction controller (ROM system) is switched on and a group of control rods is inserted into the core at normal speed to reach a lower power level in the reactor;
- up to reactor scram setpoints (AZ) the reactor protection system is actuated, all control rods are simultaneously inserted into the core at full speed and the reactor is tripped.

5.6.4.2 Reactor protection system and other safety systems

Monitoring and identification of NPP safety status

The monitoring and control system provides an automated diagnosis of the state and the operating conditions of the NPP. Monitoring and presentation of information is carried out on the reactor coolant system, on the containment, on all the systems important for safety under all operating conditions of the NPP. Remote control of these systems is possible. The operating personnel monitors the NPP systems as well as the parameters defining the NPP safe status in accordance with the service manuals from the main control room (MCR). Engineered features of the on-line diagnosis system are provided to give a possibility for an operator to form a correct estimate of the plant state, and to take necessary measures during and after an accident.

Facilities and presentation of information important for safety

The facilities for presenting information, including displays and instrumentation for monitoring safety systems, ensure:

- indication of control rod position;
- monitoring of neutron flux during operation and refuelling;
- monitoring of the level of radioactive contamination of the ground.

The control of the following systems is ensured:

- emergency protection of reactor;
- confining system;
- safety systems;
- state of protected equipment.

The automatic safety systems comprise:

- reactor scram system
- primary overpressure protection system
- emergency core cooling system
- system of passive heat removal from the reactor plant
- system of passive heat removal from the containment
- system of quick-acting isolation valves in steamlines
- secondary overpressure protection system
- diesel-generators
- system of reliable direct current power supply.

Reactor scram system

The reactor scram system provides a reliable switch-off of the electric power supply to the control rod drives, causing the scram rods to drop into the core. In this case, the disappearance of the signal of the original cause does not stop the initial action of the emergency protection.

The automatic shutdown system is designed for generating and executing the commands for limiting or decreasing reactor power, or for reactor shutdown when any accidents occur as a result of reactor plant failures or of operator errors. The following types of preventive and emergency commands are considered:

- sequential movement of control rod groups downwards with nominal speed up to the disappearance of the emergency signal (first type of preventive protection);
- prohibition of control rod upwards movement (second type of preventive protection);

- drop of all control rods to the lowest position (emergency shutdown);
- drop of one group of control rods downwards to the lowest position (unit accelerated preventive shutdown).

The scram rods are actuated by de-energizing the control rod drive mechanisms. Two sets of instrumentation for generating the commands for the reactor scram are provided. The sets operate in parallel and use an "or" logic. The signals for actuating the scram are generated using a "2 out of 3" logic in any set. The following physical parameters are used to generate the above mentioned commands:

- decrease of reactor period
- increase of neutron flux
- decrease of margin to saturation temperature in any hot leg
- increase of coolant temperature in any hot leg
- decrease of pressure differential over the reactor coolant pumps
- de-energization of several reactor coolant pumps
- decrease of pressure in the reactor
- increase of pressure in the reactor
- decrease of water level in the SG
- increase of pressure in the containment.

These parameters lead to the required decrease of reactor power, meeting the design criteria under all design conditions. Automatic disconnection of power governors and interlocking all operator's actions on control rods occur under scram.

To eliminate the consequences of severe (beyond-design) accidents with the control rods assumed failed, injection of a boric acid solution with a boron concentration of 16 g/l into the hot legs is provided from ECCS tanks by two independent systems. The pumps are started up in 15 s after the voltage is applied.

In case of a main control room (MCR) failure, for example during a fire, the reserve control room (RCR) is used to provide:

- reactor shutdown;
- monitoring of subcriticality;
- reactor cooldown;
- putting into operation of confining systems.

The possibility of control of the systems important for safety is retained from the RCR. Autonomous habitability under conditions of unavailability of regular ventilation systems is provided for the reserve control room for design events including safe shutdown earthquake (SSE) and, connected with it, fire and other site damage.

Access to the RCR is provided by an admittance check system. The RCR ensures the life support when the normal ventilation systems are de-energized in the case of anticipated impacts including SSE with accompanying fires and destructions. Local control panels which do not require interaction with the MCR and the RCR are provided. Their existence, in a number of cases, is determined by considerations of nuclear power plant layout.

5.6.5 Electrical systems

5.6.5.1 Operational power supply systems

The schematic diagram of electric connections of the unit is presented in Figure 5.6-9. The normal and the emergency electric power supply system consists of two trains of 100% capacity with each channel being divided into three groups considering reliability aspects and the time interval of loss of electric power (fraction of second; time to be specified by safety conditions for various groups of equipment without increased requirements).

Start-up of the two diesel-generators, one for each channel of reliable electric power, and to be put into operation in the case of failure of the main and the reserve grid connections, is carried out in a time not exceeding 15 s from the moment of generation of a command for start up.

DC electric power supply of the reactor control and protection system is ensured by accumulator batteries (in each channel) designed for a discharge over 24 hours. Electric power from the accumulator batteries during a station blackout is provided for both the MCR and the RCR in full measure.

5.6.5.2 Safety-related systems

Diesel generators and system of reliable DC power supply

Two physically separated diesel generators provide power to the safety related systems for 2 days using its own inventory of fuel, and for an unlimited time when fuel is delivered from the outside. The system of reliable direct current power supply comprising storage batteries, provides the power supply to electromagnetic circuits for the operation of automatic safety systems as well as for recording of necessary parameters during 24 hours.

5.6.6 Safety concept

5.6.6.1 Safety requirements and design philosophy

No information provided.

Deterministic design basis

No information provided.

Risk reduction

No information provided.

External and internal hazards

No information provided.

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

Safety systems configuration

No information provided.

Emergency core cooling system

The emergency core cooling system (ECCS) comprises the following complex of automatically initiated subsystems:

- subsystem of hydrotanks with nitrogen under pressure
- subsystem of hydrotanks under atmospheric pressure
- subsystem of deliberate emergency depressurization

AC power supply is not needed for fulfillment of the ECCS functions; the emergency core cooling system is based on the principle of passive operation providing for long-term residual heat removal in LOCA accidents accompanied by a station blackout. In the first stage of the accident, the nitrogen-pressurized hydrotanks will be actuated. When these are empty, the tanks holding cooling water under atmospheric pressure begin to operate. They are arranged to have a hydrostatic head of 20 m of water column in relation to the core. A pool formed around the reactor provides for residual heat removal from the core due to natural circulation. Active elements of the system needed for the function of emergency heat removal are provided with electric power from storage batteries.

System of passive heat removal from the reactor plant

The WWER-640 design incorporates a system of passive residual heat removal (PHRS) from the reactor plant. The design basis for this system is that in a station blackout situation, including loss of emergency power supply, the removal of residual heat shall be ensured for 24 hours, without damage to the reactor core and to the primary circuit boundary. It consists of four independent trains; each of them comprising a steam-water heat exchanger, piping for steam supply and condensate return, and battery-operated valves. The heat exchangers are installed in a tank of chemically demineralized water. The heat exchangers are connected to the secondary side of the steam generators, one heat exchanger to each loop, in such a way that the steam from the steam generator will flow to the heat exchanger where it condenses, transferring its heat to the water. The condensate will flow back to the steam generator. Coolant motion occurs owing to natural circulation.

System of passive heat removal from the containment

This system removes heat from the containment in accidents caused by loss-of-sealing of the primary circuit (LOCA). The main functions of the system are:

- emergency isolation of service lines passing through the containment and not pertaining to systems coping with the accident,
- removal of the heat released in the course of the accident into the containment,
- retention of radioactive products released into the containment,
- fixing of the iodine released into the containment atmosphere.

For heat removal there are provided coolers, storage tanks of cooling water and connecting pipelines in the system. Steam released to the containment condenses on the heat exchange surface of the cooler giving heat to the water of a storage tank in natural circulation. For iodine fixing, tanks with the iodine fixing solution are connected via quick-acting valves with the collector of a sprinkler device as the pressure in the containment increases. The system does not require any alternating current power supply.

Primary overpressure protection system

This system comprises two safety valves intended for the discharge of steam or a steam-water mixture from the pressurizer as its pressure increases above the permissible limit, as well as a subsystem for receiving the steam-water mixture. This subsystem involves a bubbler and pipelines connecting it with the outlets of the safety valves.

Secondary overpressure protection system.

This system prevents the secondary pressure to increase above the permissible level of 115% of the secondary design pressure. It incorporates quick-acting steam dump valves and steam generator safety valves.

Quick-acting isolation valves in steam lines

The system of quick-acting isolation valves in the steam lines causes closing at:

- level increase in the SGs above the permissible one
- increase of radioactivity in the SGs above the permissible one
- appearance of signals indicating a steamline rupture

The system provides for:

- protection of the turbine from steam of high humidity
- prevention of radioactivity release from SGs
- restriction of steam blowing down during rupture of the secondary circuit.

5.6.6.3 Severe accidents (beyond design basis accidents)

Severe accident mitigation strategy

No information provided.

Severe accident prevention and mitigation features

No information provided.

5.6.7 Plant layout

5.6.7.1 Buildings and structures, including plot plan

Design requirements
Design bases

Seismic effects. The design is performed taking into account two levels of seismicity: an operating basis earthquake (OBE) of magnitude 7 on the MSK-64 scale and a safe shutdown earthquake (SSE) of magnitude 8 on the MSK-64 scale.

The reactor plant equipment is designed for seismic effects proceeding from the following conditions. During operating basis earthquake, normal operation of the reactor plant is to be provided. During the safe shutdown earthquake, reactor and plant shutdown, cooling and fuel discharge are to be provided. All civil structures, process and electrotechnical equipment, pipelines, instrumentation, and so on, are divided into three seismic categories depending upon the degree of responsibility for safety ensurance during seismic events and on the serviceability after the earthquake. Components and systems of category 1 (the highest) shall fulfill their safety functions during and after an earthquake of SSE intensity. After an OBE, serviceability is maintained.

Seismic category 1 equipment includes:

- systems for normal operation, failure of which during an SSE may result in radioactivity releases causing excessive population doses in comparison with the specified values for SSE conditions;
- safety systems for keeping the reactor in a subcritical state, for ensuring emergency heat removal and for confinement of radioactive products;
- structures and equipment which could impair above functions as a consequence of an SSE.

The design considers the possibility of using special seismic isolators located under the base plate.

Loads due to wind, hurricane and tornado: The external wind load for the first category buildings and constructions is assumed to amount to 0.9 kPa, corresponding to a hurricane wind speed of 38 m/s. Effects of a tornado (sandstorm) are taken into account in the design for the first category of buildings and structures with the following characteristics and physical parameters:

- maximum horizontal speed of rotation of the tornado wall is 60 m/s;
- translational motion speed of the tornado is 15 m/s;
- tornado radius is 50 m;
- maximum wind front pressure is 3.5 kPa;
- the pressure differential between the center and the periphery of a whirlwind is 4.4 kPa;
- impact of missiles carried away by a whirlwind with a speed of 20 m/s are considered.

External industrial hazards and airplane crash: The design is based on:

- front pressure of the assumed explosion shock wave is 30 kPa;
- duration of the compression phase is 1s;
- direction of propagation is horizontal;
- impact of a plane with 5.7 t mass at a speed of 100 m/s is considered.

Loads on the inner protective steel shell: Design parameters are:

- effect of maximum excess pressure is 0.4 MPa and the maximum temperature is 150°C taking into account design and beyond-design basis accidents;
- earthquakes as explained in Section 2.10.1.;
- loads during approval tests are 0.46 MPa for pressure and 20°C for temperature.

The size and the energy of missiles originating inside the containment are determined in the design with regard for the "leak before break" concept. Mechanical effects of such missiles, and of steam-water jets on the inner shell are mitigated by means of protective shields.

Containment protection against internal pressure: The leakness of the inner shell at a maximum pressure equal to 0.5MPa is not allowed to be more than 0.2% of volume per day. During design accidents the confining safety systems ensure confinement of radioactive materials inside the protective shell, heat removal from the hermetic shell, and control and suppression of hydrogen.

Beyond-design accidents: The following confining safety systems are provided for beyond-design accidents with severe core damage:

- system of emergency gas removal from the primary circuit;
- system of control and suppression of hydrogen (hydrogen igniters);
- system of discharge and decontamination of confinement medium (2×100%), ensuring the filtered release.

Physical separation aspects, and Radiation protection aspects (accessibility, shielding, ventilation):

No information provided.

5.6.7.2 Reactor building

No information provided.

5.6.7.3 Containment

No information provided.

Containment integrity

No information provided.

Corium retention

No information provided.

5.6.7.4 Turbine building

No information provided.

5.6.7.5 Other buildings

Safeguard and fuel buildings

No information provided.

5.6.8 Technical data

General plant data		
Power plant output, gross	640	MWe
Power plant output, net		MWe
Reactor thermal output	1 800	MWt
Power plant efficiency, net	%	
Cooling water temperature	°C	
Nuclear steam supply system		
Number of coolant loops	4	
Primary circuit volume, including pressuriser		m^3
Steam flow rate at nominal conditions [4x894 t/h]	993.3	kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	- /7.06	°C/MPa
Feedwater temperature/pressure	223/	°C/MPa
Reactor coolant system		
Primary coolant flow rate [14.91m^3/s]	11 000	kg/s
Reactor operating pressure 15.7	MPa	
Coolant inlet temperature, at RPV inlet	293.9	°C
Coolant outlet temperature, at RPV outlet	323.3	°C
Mean temperature rise across core		°C
Reactor core		
Active core height	3.53	m
Equivalent core diameter	3.16	m
Heat transfer surface in the core	4 957	m^2
Fuel weight {UO_2]	68.64	t
Average linear heat rate	11.34	kW/m
Average fuel power density	29.75	kW/kg U
Average core power density (volumetric)	65.4	kW/l
Thermal heat flux, F_q	363.1	kW/m^2
Enthalpy rise, F_H		
Fuel material	Sintered UO_2	
Fuel assembly total length	4 670	mm
Rod array	triangle	
Number of fuel assemblies	163	
Number of fuel rods/assembly	294	
Number of guide tubes for control rods/instr.	18/1	
Number of spacers		
Enrichment (range) of first core	3.0	Wt%
Enrichment of reload fuel at equilibrium core	3.6	Wt%
Operating cycle length (fuel cycle length)	10	months
Average discharge burnup of fuel	40 400	MWd/t
Cladding tube material	zirconium alloy	
Cladding tube wall thickness	0.61	mm
Outer diameter of fuel rods	9.1	mm
Overall weight of assembly	kg	
Active length of fuel rods	3 530	mm
Burnable absorber, strategy/material		
Number of control rods	121	
Absorber rods per control assembly	18	
Absorber material	B_4C	
Drive mechanism	Magnetic jack	
Positioning rate		mm/min
Soluble neutron absorber	boron acid	
Reactor pressure vessel		
Cylindrical shell inner diameter	4 070	mm
Wall thickness of cylindrical shell	190	mm
Total height	19 100	mm
Base material: cylindrical shell	15Kh2NMFA	
RPV head		
Design pressure/temperature	17.65/350	MPa/°C
Transport weight (lower part) [inc. head]	302	t
RPV head		t

Steam generators

Type	horizontal, U-tube heat exchanger	
Number	4	
Heat transfer surface	4 286	m^2
Number of heat exchanger tubes	8442	
Tube dimensions	16×1.5	mm
Maximum outer diameter	4 100	mm
Total height	7 300	mm
Transport weight	300	t
Shell and tube sheet material	10GN2MFA/0Kh18N10T	
Tube material	08Kh18N10T	

Reactor coolant pump

Type	Single-stage, centrifugal pump	
Number	4	
Design pressure/temperature	17.6/350	MPa/°C
Design flow rate (at operating conditions) [3.728m^3/s]	2 750	kg/s
Pump head		
Power demand at coupling, cold/hot	2 600/1 800	kW
Pump casing material	Stainless steel	
Pump speed	1500	rpm

Pressuriser

Total volume	79	m^3
Steam volume: full power/zero power	24	m^3
Design pressure/temperature	17.65/350	MPa/°C
Heating power of the heater rods	2 520	kW
Number of heater rods	28	
Inner diameter	3 000	mm
Total height	13 000	mm
Material	10GN2MFA	
Transport weight	214	t

Pressuriser relief tank

Total volume	m^3	
Design pressure/temperature	MPa/°C	
Inner diameter (vessel)	mm	
Total height	mm	
Material		
Transport weight		t

Primary containment

Type	Dry, double wall	
Overall form (spherical/cyl.)	cylindrical, in steel/concrete	
Dimensions (diameter/height)	41/61.5	m
Free volume, gross	60 000	m^3
Design pressure/temperature (DBEs)	⁻ 500/	kPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate, inner wall	0.1	vol%/24 h
Is secondary containment provided?	yes, space between the walls	

Reactor auxiliary systems

Reactor water cleanup,	capacity	kg/s
	filter type	
Residual heat removal,	at high pressure	kg/s
	at low pressure	kg/s
Coolant injection,	at high pressure	kg/s
	at low pressure	kg/s

Power supply systems

Main transformer,	rated voltage	kV
	rated capacity	MVA
Plant transformers,	rated voltage	kV
	rated capacity	MVA
Start-up transformer	rated voltage	kV
	rated capacity	MVA
Medium voltage busbars (6 kV or 10 kV)		
Number of low voltage busbar systems		
Standby diesel generating units: number		
	rated power	MW
Number of diesel-backed busbar systems		
Voltage level of these		V ac
Number of DC distributions		
Voltage level of these		V dc
Number of battery-backed busbar systems		

Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)		
Number of turbine sections per unit (e.g. HP/LP/LP)		
Turbine speed		rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/°C

Generator

Type		
Rated power		MVA
Active power		MW
Voltage		kV
Frequency		Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type		
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate		m^3/s
Cooling water temperature		°C
Condenser pressure		hPa

Condensate pumps

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

Condensate clean-up system

Full flow/part flow		
Filter type		

Feedwater tank

Volume		m^3
Pressure/temperature		MPa/°C

Feedwater pumps

Number		
Flow rate		kg/s
Pump head		
Feedwater temperature		°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages		
Redundancies		

5.6.9 Project status and planned schedule

Construction of a first pilot plant at the Sosnovy Bor site, the Leningrad nuclear power station site, outside St. Petersburg is under consideration; planning may start within this year. This pilot plant will be followed by a number of units at the Kola nuclear power station site.

References

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[3] WWER-640 description in "Review of design approaches of advanced pressurized LWRs." Report of a Technical Committee Meeting and Workshop held in Moscow, Russian Federation, 10-13 May 1994, IAEA-TECDOC-861, 1996.

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CHAPTER 6. DESIGN CONCEPTS REQUIRING SUBSTANTIAL DEVELOPMENT

6.1 ISIS, ANSALDO, ITALY

6.1.1 Introduction

The ISIS (Inherently Safe Immersed System) design has been conceived and developed by ANSALDO in recent years beginning from 1987, as an innovative reactor with easily understandable inherent safety characteristics.

Basically, it is an integral type PWR which is completely immersed in a large pool of cold borated water. It builds upon the density lock concept originally proposed by ABB Atom for the PIUS plant and embodies revolutionary ideas for enhanced passive safety. The ISIS reactor components, on the other hand, are mainly based on proven technology derived from ANSALDO experience in the field of both light water reactors (LWRs) and liquid metal fast breeder reactors (LMFBRs).

The ISIS design addresses the following main targets:

- ensured safety by:
  - the engineered configuration providing defence, primarily by prevention, against accidents involving severe core damage;
  - self-depressurization of the primary system, scrubbing effect of the pools and absence of significant reactor containment pressurization under all accident conditions drastically reduce potential releases of radioactive material to the environment.
- improved economics by a reduced investment cost and a short construction schedule arising from the compact layout of the reactor module, made possible by the integrated reactor configuration, and the elimination of active safety systems, which are no longer needed.
- flexibility to fit demand of power producers due to:
  - the reduced unit power (200 MWe) of the ISIS reactor module permits ISIS deployment for local electrical grids;
  - the integrated reactor configuration submerged in the large pool of water makes it possible to arrange a multi-module nuclear power plant with a power level suitable for large interconnected electrical grids.
- increased range of applicability due to the outstanding passive safety features of the ISIS reactor which should permit installation wherever the combined function of electric energy generation and district heating or water desalinisation is required.

6.1.2 Description of the nuclear systems

6.1.2.1 Primary circuit and its main characteristics

The primary system of the ISIS reactor is of the integral type (Figure 6.1-1), with the steam generator (SG) unit housed in the reactor pressure vessel (RPV), to which feedwater and steam piping are connected. Within the RPV, an inner vessel provided with wet metallic insulation separates the circulating low-boron content primary water from the surrounding highly borated cold water.

The hot and cold water masses are hydraulically connected at the bottom and at the top of the inner vessel by means of open-ended, vertical, tube bundles referred to in the following as the lower and upper density locks. The inner vessel houses the core, the steam generator and the reactor coolant pumps.

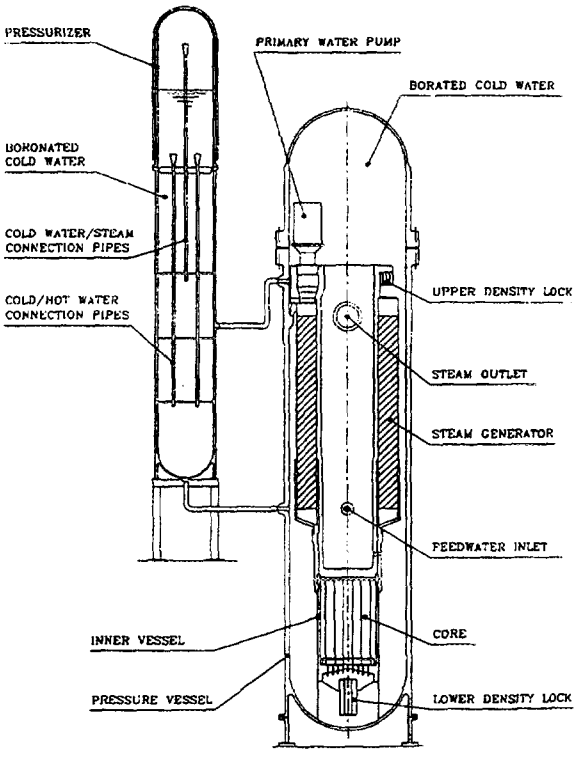


Figure 6.1-1 The ISIS reactor module

The complete immersion of the reactor coolant pressure boundary (RCPB) in a large pool of borated cold water is the outstanding feature of the ISIS design. The RCPB is for each module made up of a reactor vessel and a separated pressurizer with interconnecting piping.

During normal operation, the heat generated in the core is transferred to the steam generator via the primary coolant that is circulated by the reactor coolant pumps. These pumps are located in the "hot leg" at the top of the inner vessel. In case of unavailability of this heat transfer route, the water of the intermediate plenum (approximately 300 cubic meters per reactor module) that surrounds the inner vessel, will enter the primary circuit, mix with the primary coolant, shut down the reactor and cool the core in natural circulation.

The same water mixing process will heat up the intermediate plenum water and the pressure vessel and activate the natural heat transfer route towards the reactor pool, which contains approximately 6 000 cubic meters of cold water. The water inventory in the reactor pool is large enough to ensure that the water temperature will remain below the boiling point at atmospheric pressure after removal of the decay heat for about a week.

The water of the pool is cooled by two cooling loops in natural circulation, each provided with an air cooler dissipating the heat to a chimney outside the containment. The coolers are sized to reject approximately 2 MW to the external atmosphere and thereby capable of keeping the pool water temperature below 80°C, for an unlimited time.

6.1.2.2 Reactor core and fuel design

The reactor core consists of 69 typical (17×17) PWR fuel assemblies with a reduced active length (2.92 m) to limit the pressure losses and with a low power density (70 kW/l) for increased design margins. Soluble boron and burnable poisons are used for shutdown and fuel burnup reactivity control. The use of burnable poisons for partial reactivity control results in a reduced need of boron concentration in the primary circuit during power operation, and assures a non-positive moderator temperature coefficient at any operating condition.

6.1.2.3 Fuel handling and transfer systems

The ISIS fuel handling and storage systems are conventional. Spent fuel removed from the reactor vessel is temporarily stored under water. The spent fuel racks are located inside the spent fuel storage pool under sufficient water to provide radiological shielding. The pool can store up to 3 reactor cores.

6.1.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel (RPV) is of cylindrical shape with hemispherical heads. The main openings of the reactor vessel are the nozzles for the steam and feedwater lines and the two pipe connections to the pressurizer. The construction material is low-alloy carbon steel, internally lined with austenitic stainless steel. The RPV is a large component (4.9 m in diameter and 26.5 m high) which has several functions: it contains the whole primary system and the intermediate water plenum and operates as heat exchanger to remove the decay heat to the reactor pool. The intermediate water plenum limits the total neutron fluence of the RPV to less than 10^{15} n/cm^2 over its life time.

Reactor internals

The above core structure which is shaped like a flat-bottom cylindrical glass, provides the support for the core instrumentation and forms the inner wall of the annular riser of the primary water. The structure is open at the top. The water within it is part of the intermediate plenum, and this helps to limit the primary water inventory in the reactor module to a minimum. The above core structure is flanged to and suspended from the top of the inner vessel for easy removal to allow standard fuel handling.

Steam generator

The steam generator (SG) features an annular tube bundle with helical tubing. The steam is generated on the tube side. The feedwater piping is connected to two feedwater headers, located symmetrically inside the reactor vessel within a calm zone, each provided with two tube plates laid out vertically. The tubes depart circumferentially from the tube plates.

A similar arrangement is provided at the top for the connections of the two steam headers. The vertical arrangement of the tube plates aims at preventing crud deposition at the tube-to-tube plate connections, where corrosion can be predicted to occur. The pressure on the outside of the tube is higher than the inner pressure, a reversed situation compared with conventional steam generators, and the compression forces reduce the probability of flaw growth in the tubes.

The steam pressure is 4.6 MPa (at the SG outlet) with about 30°C of superheating. The secondary side water inventory inside the tube bundle is significantly small in comparison to the primary water inventory (3.5 tonnes of secondary water inventory against 20 tonnes of primary water in normal operating condition) which limits the feedback effects on the core of steam line break accidents.

The normal decay heat removal function of the steam and feedwater systems is not safety-related. Closure of the isolation valves in both steam and feedwater lines allows a complete separation of the nuclear steam supply system (NSSS) from the non-safety-grade balance of plant (BOP). The system is designed for the maximum expected pressure, and no need for pressure relief is anticipated. Relief valves are nonetheless provided in accordance with current codes and regulations.

Pressurizer

The pressurizer is a pressure vessel of slim cylindrical shape with hemispherical heads. The pressure control function is carried out in the upper part, which is externally insulated to limit heat losses from the steam and hot water plena.

The remaining bottom part contains cold water, hydraulically connected to the upper hot water plenum by means of a number of vertical pipes.

The function of the pipes is to enhance mixing of the hot water with the cold water, in case of water flow towards the reactor vessel during transients.

The pressurizer and reactor vessel are connected hydraulically by means of two pipes, one from the top and one from the bottom of the cold water plenum of the pressurizer to the top and mid-height of the cold water mass in the reactor pressure vessel, respectively, in order to create a common mass. The solution with two connection levels was chosen to enable natural coolant circulation in case of temperature differences between the cold water masses. If the normal decay heat removal route (i.e., the active steam/feedwater system) is lost, the un-insulated wall portion of the pressurizer would help the decay heat removal by conducting heat to the reactor pool.

Water movement between the two vessels, in reality parts of a common cold water plenum, does not significantly contribute to the thermal loadings on the reactor coolant pressure boundary during transients.

Reactor coolant pumps

The two reactor coolant pumps (RCP) are of the glandless, wet winding type and are fully enclosed within the reactor vessel. The pump motor is cooled by the water of the intermediate plenum.

A variable speed operation is required to control the "hot-cold" interface in the lower density lock. The variable frequency electric supply is provided by two generators driven by a common variable-speed turbo-coupling with a flywheel of high mechanical inertia (about 5000 kg m^2).

Main coolant lines

ISIS is an integral reactor design and the primary coolant circulates within the reactor pressure vessel; there are no external main coolant lines.

6.1.2.5 Reactor auxiliary systems

A flow diagram for the reactor auxiliary systems is shown in Figure 6.1-2.

Chemical and volume control system

The chemical and volume control system (CVCS) does not belong to the safety-grade systems. It is made up of two independent system trains; both are used at reactor start-up, but only one is needed at normal operation of the reactor.

The main functions of the CVCS are:

- to control the chemistry of the primary water;
- to control the chemistry of the intermediate plenum water;
- to control the hot/cold interface level of the upper density lock under normal operation;
- to control the water level of the pressurizer; and
- to control the boron concentration of the primary water. The concentration is controlled by boron transfer between the primary and the intermediate plenum water without buffering vessels, in order to keep the overall boron content inside the reactor module constant and sufficient to bring the reactor to cold shutdown when the water of the intermediate plenum enters the primary loop and mixes with the primary coolant.

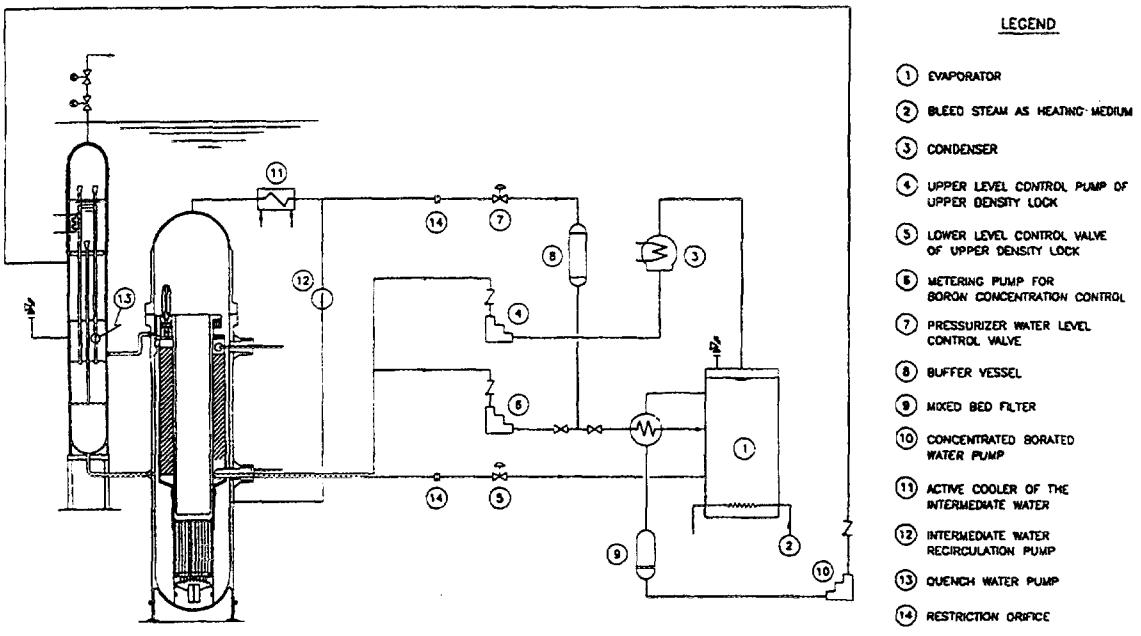


Figure 6.1-2 ISIS - Reactor auxiliary systems flow diagram

6.1.2.6 Operating characteristics

ISIS can be operated as a base load power plant at any power level between about 10% and 100%, and it can accommodate power changes of up to 10% per minute. It is also capable of automatic load following operation in 100-50-100% cycles.

It has been designed to withstand a number of operational transients and occurrences, e.g., a full load rejection, without tripping of the reactor.

6.1.3 Description of turbine generator plant system

6.1.3.1 Turbine generator plant

The steam generated in the steam generators is supplied to the high-pressure turbine via the main steam header. When leaving the high pressure turbine, the steam passes through two combined moisture separators/reheaters prior to entering the three low-pressure turbines.

This arrangement refers to a three-reactor module plant with one 600 MWe turbogenerator set. (An alternative configuration features a 200 MWe turbogenerator set for each module which, in that case, includes one high pressure turbine, one moisture separator/reheater and one low pressure turbine).

6.1.3.2 Condensate and feedwater systems

The main condenser is of a three shell type: one shell is located at each low-pressure stage turbine exhaust. The tube bundle is perpendicular to the turbine axis.

The main condenser is of single flow type in case of sea or river circulating water and of double flow type in case of cooling tower circulating water.

Two 50% capacity condensate pumps operate in parallel during normal operation. A third 50% capacity condensate pump is provided and maintained in standby for automatic starting if required. Each pump is electric-motor driven, vertical and installed at an elevation that allows operation at low condensate level in the main condenser hotwell.

The feedwater system consists of three 50% electric-motor driven main feed pumps, taking suction from the deaerator. Feedwater flow control is achieved by adjusting the main feed pump speed and the feedwater flow control valves.

Consistently with the relatively low feedwater temperature (120°C), no high pressure heaters are provided.

No specific auxiliary feedwater system is included in the design; residual heat can be removed by a number of non safety systems:

- the auxiliary startup system (heat removing heat from the secondary side of the steam generator)
- the chemistry and volume control system (removing heat from the primary system)
- the intermediate pool cooling system.

6.1.3.3 Auxiliary systems

The primary water chemistry is controlled by the chemistry and volume control system (CVCS); additional systems are provided to control the reactor pool chemistry and the secondary side water chemistry which needs to be addressed with particular emphasis in order to limit the deposits on the inner surface of the once-through steam generator tubes in the boiling region.

Provision for treatment, handling and disposal or storage of gaseous, liquid and solid waste are included. For the multi-modular design arrangement a common radwaste building is provided.

6.1.4 Instrumentation and control systems

6.1.4.1 Design concepts including control room

The plant control scheme is basically a "turbine follows reactor", with a certain degree of co-ordination to speed up the overall response.

The approach used for the reactor power control of the ISIS design is to operate at constant core outlet temperature in order to minimize the control duty of the variable speed reactor coolant pumps, which have to maintain the equilibrium interface in the lower density lock. The reactor does not have any control rods, and the reactor power level is controlled by the boron content and the temperature of the moderator/coolant. The steam generator has a quite low heat capacity, and a change in the feedwater flow rate will rapidly yield a change in the core coolant inlet temperature. Rapid changes in the reactor power level are therefore accomplished by inserting or extracting core reactivity through a lowering or an increase in the core inlet temperature by control of the feedwater flow rate.

Deviations in the core outlet temperature from the set value are controlled by adjusting the boron concentration in the primary coolant, by insertion or removal of boron via the CVCS.

The steam line pressure is controlled by the steam pressure control system by means of the steam flow control valves.

Thermal expansion or contraction of the primary water is accommodated by means of withdrawal or injection of water from or into the primary system, without any water flow through the density locks.

The compensating function is carried out by the upper density lock control system, while the control system of the lower density lock affects the speed of the reactor coolant pumps to keep the equilibrium interface in an appropriate position.

The basic scheme of the ISIS control systems is shown in Figure 6.1-3.

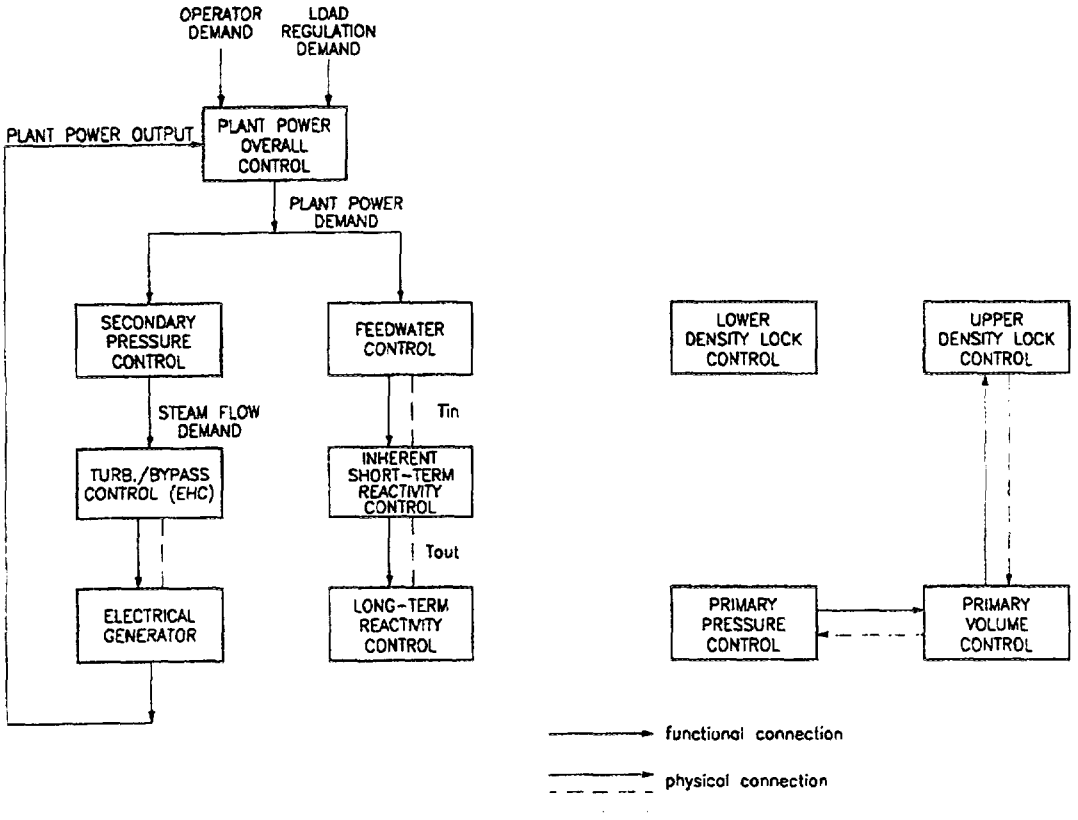


Figure 6.1-3 ISIS - Conceptual plant control scheme

An advanced control room, with a wall panel information station and work positions for operators and supervisor is foreseen, taking into account the most recent achievements and improvements in man-machine interface systems.

6.1.4.2 Reactor protection system and other safety systems

The reactor protection system of ISIS (characterized by the absence of control rods and by the "hot-cold" interface control in the density locks) serves to initiate safety functions before the response of the passive functions is activated, in order to achieve milder transients. The passive functions are used as back-up to the active systems; the only active safety function needed is the isolation of the containment. Even though the reactor protection system and the safety parameter monitoring system are not strictly required to be safety-related in the sense of common understanding, but they are still designed as Class 1E equipment, utilizing proven, distributed microprocessor-based technology.

Different measurements are monitored to detect departures from the design conditions. Independent and redundant channels and the combination of a main and a second independent measuring parameter will result in a system reliability as high as achieved by the most modern safety-related systems.

The reactor protection system provides the acquisition of field signals and actuates the logic of primary pump runback, trip of the feedwater pumps and containment isolation. The remaining few auxiliary systems designed to the electrical safety class 1E are the emergency lighting , radiation monitoring and the no-break power supply system (UPS).

6.1.5 Electrical systems

6.1.5.1 Operational power supply systems

The electrical supply system is based on a "two train" approach. Only non-safety-grade diesel generators are provided in the ISIS plant. Their availability allows the reactor to be maintained in a hot shut-down state following a number of transients, hence facilitating the subsequent startup.

6.1.5.2 Safety-related systems

Not applicable for ISIS (Section 6.1.6.2).

6.1.6 Safety concept

6.1.6.1 Safety requirements and design philosophy

Emphasis has been given to prevention of core degradation accidents. The two main safety functions, reactor shutdown and decay heat removal, are performed without recourse to the usual sensor-logic-actuator chains, i.e. with no inputs of "intelligence", nor external power sources or moving mechanical parts, corresponding to Passive Components of Category B according to the definition of the IAEA-TECDOC-626. Active protective measures, aimed at preceding and preventing passive system interventions, are included in the design but not credited in the safety analysis.

The ISIS reactor response to design basis accidents, highlighting its passively safe behaviour, gives a good illustration of the design philosophy.

The always open connections between the primary circuit and the surrounding cold borated water provides for a reliable shutdown and residual heat removal, upon disturbances in the electric power supply. During normal plant operation, the natural circulation loop of the highly borated water of the intermediate plenum through the lower density lock, the core, the riser, and back to the intermediate plenum via the upper density lock, is kept inactive by the reactor coolant pumps. In the event of loss of the plants auxiliary power supply, the pumps will coast down and the natural circulation will be established resulting in reactor shutdown and providing continued core cooling.

If the heat sink is lost, the secondary side of the steam generator heats up rapidly, but without major consequences. Even in case of a spurious instantaneous isolation of the steam and feedwater lines, resulting in a rapid drop in the steam generator heat transfer capability to zero, the secondary side of the steam generator reaches thermal equilibrium with the primary side with a pressure increase up to about 11.5 MPa which is below the design pressure of the secondary system. At the same time, a heating-up of the primary water occurs with an associated decrease of nuclear power due to the negative moderator temperature reactivity feedback.

Loss of coolant accidents represent significant challenges for existing PWR plant designs, but for ISIS they are less important. Given the integrated configuration of the primary system, no significant breaks in the primary circuit are conceivable; only a loss of pressure boundary of the intermediate pool can occur. The largest pipe of this pool has a cross section of only about $0.01 m^2$, and the outflow will be quite limited. Besides, all pipe connections are located well above the top of the core.

Scoping parametric analyses have been carried out, conservatively assuming $0.05\ m^2$ large breaks located at different elevations, including the vessel bottom, and they always show the same type of accident evolution. A series of simultaneous water transfers occur: intermediate pool water flows out of the break into the reactor pool and primary water flows into the intermediate pool, through both density locks. This phase lasts as long as the depressurization produces steam bubbles in the upper and hottest zone of the primary system and brings about cavitation of the primary pumps. Then, degradation of the pump performance causes entrance of the intermediate water into the primary loop via the lower density lock with subsequent reactor scram.

A steam generator tube rupture will result in an outflow of primary water into the steam generator tubing, and this outflow is compensated by water from the pressurizer. Borated water enters the primary system from the upper density lock and shuts down the reactor, while depressurizing the primary system and hence stopping the primary water loss. The active protection system (not safety-grade) only determines an earlier termination of the primary water outflow by closing the main steam and feedwater isolation valves.

In the event of steam line break accident, there will be a sudden depressurization of the secondary water inside the steam generator tubing, yielding a cooling down of the primary water flowing through the steam generator. And eventually, a cold water plug will enter the core with a subsequent transient overpower.

This primary water cooling down is inherently limited, however, by the low tube-side water inventory (about 3,5 tonnes at nominal condition) and low steam generator flowrate (high pressure loss of the helical tubing). Therefore, the transient overpower is limited to about 30% of the nominal power. Crediting a timely primary pump trip (active, non-safety-grade system), the consequences of the transient overpower can be reduced even further.

In conclusion it is noted that the passive safety features of the ISIS reactor prevent any conceivable core degradation accident and minimize the release of radioactivity by virtue of:

- a sufficient inventory of cold water inside the reactor vessel always ensures efficient core cooling; even in LOCA events the core is cooled without any transient heat-up phase;
- the mixing of the primary water with the intermediate pool water, and possibly with the reactor pool water, provides a scrubbing effect for any radioactivity released from the fuel;
- the "pool reactor" configuration results in no containment pressurization in the event of primary water loss, and consequently no significant release of radioactivity outside the containment.

There is only one safety-grade path for heat rejection to the ultimate heat sink provided in the ISIS design. It is based on concentric shells (the hot and pressurized water of the primary system, the cold and pressurized water of the intermediate plenum, the cold water of the reactor pool with atmospheric free level and, ultimately the external atmosphere). The configuration is such that its reliability will be extremely high (practically absolute, since the loss of integrity of a pressurized shell does not jeopardize the effectiveness of the path).

In spite of the above, if hypothetical core damage is assumed:

- the cold water inside and outside the vessel bottom guarantees the corium cooling and its retention inside the vessel bottom. The water which evaporates inside the vessel condenses on the cold vessel metal surface and then drains back to the vessel bottom, thus contributing to the fuel cooling indefinitely;
- the modular approach, with a common containment building, results in a favourable ratio of hydrogen produced in comparison with the available free volume.

Deterministic design basis

ISIS is based on a deterministic approach that takes into consideration the same categories of events as used for PWRs of evolutionary design. Its innovative configuration allows, however, the elimination of some Design Basis Accident initiators and the reduction of the frequency or of the severity of the remaining accident sequences. The ISIS response to a number of design basis accidents, is shortly described below.

Loss of the station service power (LOSSP). During normal plant operation, the natural circulation of the highly borated water of the intermediate plenum through the lower density lock, the core, the riser, and back to the intermediate plenum via the upper density lock, is kept inactive by the main coolant pumps. In case of a LOSSP, the pumps coast down and natural circulation sets on, causing reactor shutdown and providing continued core cooling.

Loss of heat sink (LOHS). Even in case that both steam and feedwater lines are instantaneously isolated, so that the steam generator's heat transfer capability drops quickly to zero, the secondary side of the steam generator reaches a thermal equilibrium with the primary side with a pressure increase of about 11,5 MPa, which is below the design pressure of the secondary system. At the same time, heating-up of the primary water occurs with associated decrease of nuclear power caused by the reactivity feedback of the moderator temperature.

Loss of coolant accidents. Considering the hydraulically open configuration of the integral primary system, a loss of the reactor coolant pressure boundary may be conceived only because of a break in the boundary of the intermediate pool. The cross-section of the largest pipe of the Intermediate Pool is limited to about 0.01 m^2, with all pipe connections located at a higher level than the core.

Scoping parametric analyses carried out assuming a conservatively 0.05 m^2 wide break located at different levels, including the reactor vessel bottom, always show the same type of accident evolution.

A series of simultaneous water transfers occur. Intermediate pool water flows out of the break into the reactor pool; it is replaced by primary coolant, that flows in through both density locks, while hot and cold plena mix up in the pressurizer, starting depressurization of the pressure boundary. This phase lasts until steam bubbles occur in the upper, hotter zone of the primary system and bring about cavitation of the primary pumps. Due to the cavitation, intermediate pool water then enters the primary loop via the lower density lock and shuts down the reactor.

Steam generator tube rupture (SGTR). Outflow of primary water, that enters the lower-pressure steam generator through an assumed break in the tubing, draws intermediate pool water into the primary system via the upper density lock and into the reactor vessel from the pressurizer.

The cold, borated water entering the primary system shuts down the reactor, depressurizes the primary system and stops any further primary water loss. The intervention of the (non-safety-grade, but 1E-classified) protection system is not required in this passive-only process, but would bring about, however, an earlier termination of the primary water outflow by the closure of the main steam and feedwater isolation valves (backed up by the trip of the feedwater pumps, to eliminate any further accident evolution).

Steam line break accident (SLBA). The sudden depressurization of the secondary water inside the steam generator tubing following a pipe break cools down the primary coolant flowing through the steam generator. A cold primary coolant plug flow eventually enters the core with a consequent transient over-power (TOP).

Primary water cooling is inherently limited, however, by the low tube-side water inventory (about 3.5 tonnes at nominal condition) and flow rate (because of the high pressure loss in the helical tubing). The TOP is therefore limited to about 30% of the nominal power. Crediting a timely primary pump trip by intervention of the protection system, the consequences of the TOP could be even further reduced.

Risk reduction

The ISIS configuration that is so effective in coping with design basis accidents (DBAs), offers also protection against (and mitigation of the consequences of) postulated Beyond DBA situations. Probability figures are not available yet, since a full-scope or even a level 1 PSA would be meaningless at this stage of the design work, mainly due to lack of reliable data for modelling of passive components and structures. Considering its innovative safety features and based on the investigations performed so far, however, the probabilities of core damage or of large radioactivity release to the environment can be predicted to be extremely low for ISIS.

External and internal hazards

External and internal hazards have not yet been addressed extensively for ISIS, owing to the early stage of the overall plant design. The seismic behaviour of the nuclear island should be generally favourable, however, because of the selected deep embedment into the soil and the low elevation of its center of

gravity. Given lack of seismic site data, the spectrum of the US NRC Regulatory Guide 1.60, with an SSE level of 0.3g, has been adopted for the design of the free-standing reactor pool of the general ISIS plant, which rests on a common foundation mat, but is structurally decoupled from the reactor building. The above-ground elevation of the reactor pool should offer adequate defense against external flooding. The occurrence of a fire, finally, could be the cause of accident initiation, but would, by no means, lead to degradation of the interventions essential to safety, because these interventions are carried out passively by systems immersed in water. The same deterministic design basis and risk reduction approach will be applied when considering the remaining hazards in the detailed design phase of the plant.

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

As pointed out in the design philosophy, Section 6.1.6.1, the two safety functions of reactor shutdown and decay heat removal are assured by passive systems or inherent features. Active systems intended for protection are provided as well, but their intervention is not necessary for safety, because they are always backed up by the passive systems. The only active system not backed up by a passive feature is the containment isolation for the activity confinement function.

Safety systems configuration

The safety of ISIS is largely embodied in its primary system conceptual design and layout, which is described in Section 6.1.2, nuclear system configuration, with the additional reactor pool, air-cooler and containment isolation systems, described below in Section 6.1.7.2, reactor building.

Safety injection systems

Not applicable

In-containment refuelling water storage tank

Not applicable. See duty of the reactor pool for the decay heat removal function.

Emergency feedwater system

Not applicable.

Residual heat removal system

There is no dedicated safety-related residual heat removal (RHR) system inside the ISIS module. If the operational heat transfer route via the steam generator is not available, cold water of the intermediate plenum (approx. 300 m^3 per reactor module) invariably enters the primary circuit, mixes up with the hot primary water and cools down the core in natural circulation. This water mixing process heats up the intermediate plenum water, which can be kept cooled by an active system.

If the active cooling system is not available, heating up of the pressure vessel and of the lower part of the pressurizer activates the natural heat tranfer by conduction and convection towards the reactor pool, which contains approximately 6000 m^3 of cold water. The reactor pool water inventory is large enough so that its temperature remains below the boiling point at atmospheric pressure after storing decay heat for about a week. Its temperature level is actually limited to below 80°C, for indefinite time, by the operation of two air cooler loops in natural circulation.

It can be noted that only one safety path for the residual heat removal to the ultimate heat sink is provided in the ISIS design. This path goes through the concentric barriers of the ISIS reactor and plant configuration (i.e., the mixed-up, non-pressurized water plenum within the pressure boundary; the pressure boundary itself; the cold and non pressurized reactor pool and the external atmosphere, via the air cooler loops in natural circulation, which are necessary only to guarantee heat removal for unlimited time without need for make-up water) and is never lost, provided that at least the reactor pool be available, i. e., the more straightforward among the ISIS plant's passive components.

This RHR path is therefore extremely reliable (in fact almost absolutely reliable, since even the postulated loss of integrity of the pressure boundary does not jeopardize the RHR function).

6.1.6.3 Severe accidents (Beyond design basis accidents)

Severe accident mitigation strategy

There are no credible paths leading to core degradation and to a significant release of radioactive material to the environment, according to the analyses performed so far.

Severe accident prevention and mitigation features

A hypothetically hindered coolant circulation through the core due to loosened above core structure or debris seized in the core, would possibly lead to core damage. The consequences of such an accident will still be mitigated, however, because of the safety features embodied in ISIS, as follows:

- abundant cold water inside and outside the reactor vessel bottom would cool any corium retained inside the vessel. Water evaporated inside the reactor vessel would condense on its cold inner wall and drain back, thus contributing to cooling the corium indefinitely,
- the modular approach with common containment building determines a favourable ratio of produced hydrogen to the free containment volume,
- radioactive materials released from the pressure boundary would be effectively scrubbed by the reactor pool,
- the containment atmosphere would never pressurize significantly, hence, assuming the permeability achieved by the modern average containment building, there would never be enough driving force to release a large amount of radioactive gases to the environment.

6.1.7 Plant layout

6.1.7.1 Buildings and structures, including plot plan

Each reactor module is immersed in a reactor pool, the bottom of which rests on the foundation mat of the reactor building. The reactor pool is of circular shape with an inner diameter of 17 m. The auxiliary systems for treatment of the primary water, the spent fuel storage, the fuel handling facilities and the control room are all located inside the reactor building.

In the case of a multi-module ISIS plant (up to three modules), each module has dedicated auxiliary systems. Those auxiliary systems, which treat primary water, are located in segregated room enclosures. Separated access to each potentially contaminated area is provided from the common area in the upper part of the reactor building. Handling and refuelling facilities are common systems inside the reactor building.

Design requirements

For the site-dependent internal and external hazards, the design requirements will be similar to those of modern evolutionary reactors, in particular with respect to:

- Earthquake
- Aircraft crash
- Explosion pressure wave
- Internal hazards
- Physical separation aspects
- Radiation protection aspects (accessibility, shielding, ventilation)

6.1.7.2 Reactor building

The reactor building (Figure 6.1-4) is of rectangular plot plan and designed for a relative pressure of 50 kPa (0.5 bar), which conservatively envelopes the accidental pressure increase due to the evaporation of the water of the reactor pool.

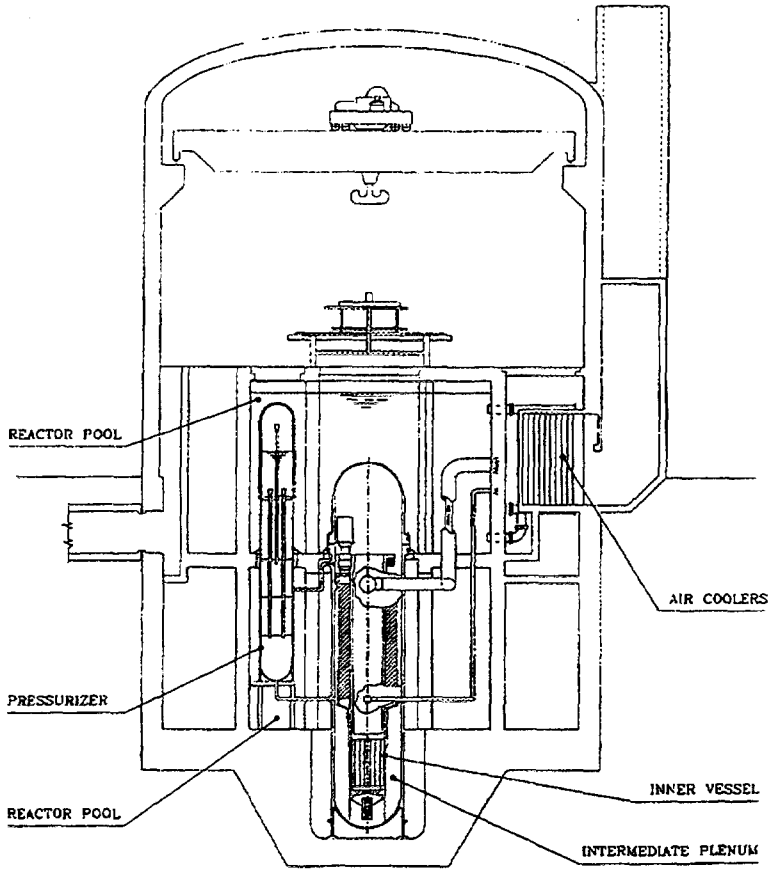


Figure 6.1-4 ISIS - The reactor building

Two reactor building sizes are foreseen: one for a single module plant and the other for a plant of three modules with shared fuel element and component handling facilities.

A relatively low cost is anticipated for the reactor building due to the low inner design pressure of the containment, no stringent requirements upon containment leaktightness, and a short construction time thanks to the building geometry that is suitable for prefabrication, and to a complete decoupling of the components installation from civil works. (The construction time has been estimated to about 20 months from start of excavation to completion of the civil structure.)

6.1.7.3 Containment

The containment of ISIS consists of the reactor building and the reactor pool. The integrity of the containment can be assumed to be the same as that of conventional reinforced-concrete structures, designed for moderate inner pressure and site-dependent hazards.

With respect to corium retention measures, reference is made to the discussion of severe accidents in Section 6.1.6.3.

6.1.7.4 Turbine building

The turbine building of ISIS is separated from the reactor building and of conventional construction and layout.

6.1.7.5 Other buildings

Remaining buildings are the radioactive waste building and the electrical and service auxiliary systems building, containing in particular the non-safety-grade diesel generator set and the variable frequency generator for the primary pumps.

6.1.8 Technical data

General plant data		
Power plant output, gross	3×205	MWe
Power plant output, net		MWe
Reactor thermal output	3×650	MWt
Power plant efficiency, net		%
Cooling water temperature		°C
Nuclear steam supply system (per module)		
Number of coolant loops	1	
Primary circuit volume, including pressuriser	75	m^3
Steam flow rate at nominal conditions		kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	290/4.6	°C/MPa
Feedwater temperature/pressure	120/5.26	°C/MPa
Reactor coolant system (per module)		
Primary coolant flow rate	2 911	kg/s
Reactor operating pressure	14.0	MPa
Coolant inlet temperature, at RPV inlet	271	°C
Coolant outlet temperature, at RPV outlet	310	°C
Mean temperature rise across core		°C
Reactor core (per module)		
Active core height	2.92	m
Equivalent core diameter	2.0	m
Heat transfer surface in the core		m^2
Fuel inventory [U or UO_2?]	24.3	t
Average linear heat rate		kW/m
Average fuel power density	26.7	kW/kg U
Average core power density (volumetric)	70	kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		
Fuel material	Sintered UO_2	
Fuel assembly total length		mm
Rod array	square, 17×17	
Number of fuel assemblies	69	
Number of fuel rods/assembly	264	
Number of control rod guide tubes		
Number of spacers		
Enrichment (range) of first core	2.0-3.0	Wt%
Enrichment of reload fuel at equilibrium core	3.5	Wt%
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel	38 000	MWd/t
Cladding tube material	Zircaloy	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods		mm
Overall weight of assembly		kg
Active length of fuel rods	2 920	mm
Burnable absorber, strategy/material	B_4C	WABA
Number of control rods	None	
Absorber rods per control assembly	N.A.	
Absorber material	N.A.	
Drive mechanism	N.A.	
Positioning rate N.A. [or mm/min or mm/s]	steps/min	
Soluble neutron absorber	Boric acid	
Reactor pressure vessel (per module)		
Cylindrical shell inner diameter	4 900	mm
Wall thickness of cylindrical shell	210	mm
Total height	26 500	mm
Base material: cylindrical shell	Carbon steel	
RPV head		
Lining	Stainless steel	
Design pressure/temperature	16.0/	MPa/°C
Transport weight (lower part) {incl. head}	782	t
RPV head		t

Steam generator (per module)

Type	Integrated, vertical, once-through, helical-tube heat exchanger	
Number	1	
Heat transfer surface	5 090	m^2
Number of heat exchanger tubes		
Tube dimensions	mm	
Maximum outer diameter		mm
Total height		mm
Transport weight		t
Shell and tube sheet material		
Tube material Inconel 690		

Reactor coolant pumps (per module)

Type	Glandless, centrifugal pump with mixed flow impeller and with wet motor	
Number	2	
Design pressure/temperature		MPa/°C
Design flow rate (at operating conditions)	1 565	kg/s
Pump head	0.2	MPa
Power demand at coupling, cold/hot [nominal]	540	kW
Pump casing material		
Pump speed		rpm

Pressuriser (per module)

Total volume	~ 72	m^3
Steam volume: full power/zero power	11/13	m^3
Design pressure/temperature	16/	MPa/°C
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter	Arranged in RPV	
Total height		mm
Material		
Transport weight		t

Pressuriser relief tank

N.A.

Total volume	m^3
Design pressure/temperature	MPa/°C
Inner diameter (vessel)	mm
Total height	mm
Material	
Transport weight	t

Primary containment

Type	Pool reactor in leaktight containment	
Overall form (spherical/cyl.)	Parallel piped with gabled dome	
Dimensions (diameter/height)	31.5×38/53	m
Free volume		m^3
Design pressure/temperature (DBEs)	5.0/80	kPa/°C
(severe accident situations)	~ /	kPa/°C
Design leakage rate	< 1	vol%/day
Is secondary containment provided?	yes, space between the walls	

Reactor auxiliary systems

Reactor water cleanup, capacity		kg/s
filter type		
Residual heat removal, at high pressure		kg/s
at low pressure		kg/s
Coolant injection, at high pressure		kg/s
at low pressure		kg/s

Power supply systems

Main transformer, rated voltage		kV
rated capacity		MVA
Plant transformers, rated voltage		kV
rated capacity		MVA
Start-up transformer rated voltage		kV
rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)	2	
Number of low voltage busbar systems	2	
Standby diesel generating units: number	2	
rated power		MW
Number of diesel-backed busbar systems	2	
Voltage level of these		V ac
Number of DC distributions 2		
Voltage level of these		V dc

Number of battery-backed busbar systems	2	
Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	reaction type	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/ 3 LP	
Turbine spee	3 000	rpm
Overall length of turbine unit	28	m
Overall width of turbine unit	8	m
HP inlet pressure/temperature	4.5/288	MPa/°C

Generator

Type	3-phase, synchronous	
Rated power	715	MVA
Active power	645	MW
Voltage	15	kV
Frequency	50	Hz
Total generator mass	280	t
Overall length of generator	10	m

Condenser

Type	three shell, single flow	
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate		m^3/s
Cooling water temperature		°C
Condenser pressure		hPa

Condensate pumps

Number	3×50%	
Flow rate		kg/s
Pump head		
Temperature		°C
Pump speed		rpm

Condensate clean-up system

Full flow/part flow	
Filter type	

Feedwater tank

Volume		m^3
Pressure/temperature		MPa/°C

Feedwater pumps

Number		3×50%
Flow rate		kg/s
Pump head		
Feedwater temperature		°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages	none
Redundancies	

6.1.9 Project status and planned schedule

The conceptual design of ISIS is completed. This activity, self-funded by ANSALDO, has included the following aspects:

- Functional design of the main components and systems with preliminary mechanical verification of the most stressed structures.
- Extensive safety analyses mainly performed with the RELAP Code, with contributions of experts from the Italian regulatory body ANPA (formerly ENEA DISP).
- Dynamic Analysis of ISIS Control System.

The current activity (1996) is focused on the study of an ISIS reactor version suitable for co-generation of electricity and heat. This activity aims at the optimization of reactor size and thermal cycle and at the study of cost estimate. The target reactor size should yield power in the range from 200 MWth to 200 MWe.

Besides the density lock concept that has been extensively tested by ABB Atom for the PIUS reactor, and although the main components are fabricated with proven technology, the ISIS reactor design development needs accompanying testing on the following topics:

- steam generator performance, (planned)
- wet-insulation performance, (to be planned)
- reactor system operational testing for concept confirmation and support of licensing. (to be planned)

Testing dealing with the steam generator is scheduled for start in the second half of the year and will include fundamental research on the flow-induced vibrations of helical-tubes of compact tube-bundle design, the selection and feasibility assurance of the tube support system and confirmation of the overall performance of the unit.

The simple plant design, the great forgivingness and the modular approach should make ISIS particularly suitable for the combined function of electric energy generation and heat distribution for water desalinisation or district heating.

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6.2 JPSR, JAERI/IHI, JAPAN

6.2.1 Introduction

The Japan Atomic Energy Research Institute (JAERI) is performing a conceptual design study on the JAERI passive safety reactor (JPSR) in an effort to enhance the safety level and reduce the manpower in operation and maintenance of next-generation light water reactors, as described in the References 1 - 5.

The JPSR is a two-loop PWR with a net electric output of 630 MWe (659 MWe gross). Its major features are:

- The heat generation rate in the core is physically matching the actual heat removal rate in the nuclear steam supply system (NSSS). This is accomplished by adopting a soluble boron free design to increase the reactivity change due to changes in moderator density. Compensation of excess reactivity without a chemical shim control system requires additional control rods, but an increased number of penetration holes through the reactor pressure vessel head for control rod drive mechanisms (CRDMs) is difficult to accommodate. Therefore, in-vessel CRDM units have been selected for manoeuvring of the rods.
- A large-volume pressurizer and passive residual heat removal (RHR) systems are adopted to retain coolant in the primary system.
- Canned-motor pumps are adopted as primary coolant circulation pumps to eliminate the pump-seal-water feed system.
- Passive safety injection systems are adopted.

Based on these features, auxiliary safety systems such as emergency diesel generators, pumps, valves, and the chemical and volume control system (CVCS) can be significantly simplified. As a specific result of the features, the reactor power can be automatically controlled by controlling the feedwater flow rate to the steam generators. Thus, the manpower in operation and maintenance can also be significantly reduced.

6.2.2 Description of the nuclear systems

6.2.2.1 Primary circuit and its main characteristics

The primary circuit of the JPSR which is outlined in Figure 6.2-1, has retained most of the arrangement of currently operating pressurized water reactors (PWRs).

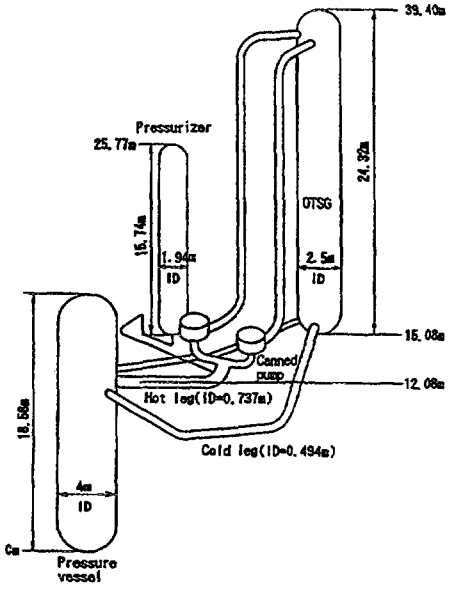


Figure 6.2-1 View of the JPSR primary coolant system

It is a two-loop PWR in which each loop consists of one hot leg and two cold leg pipes, two primary coolant pumps and one steam generator. The primary coolant pumps are installed in the hot legs to improve accessibility for maintenance.

6.2.2.2 Reactor core and fuel design

The reactor core, with a thermal power of 1 853MWt, consists of 145 fuel assemblies (Figure 6.2-2). Each assembly consists of 264 fuel rods in a square 17×17 array, similar to those of well operating PWRs. The dimensions and composition of the fuel rods are the same as those of currently operating PWRs, e.g., the pellets have a diameter of 8.19 mm, the cladding outer diameter is 9.5 mm, the fuel rod pitch is 12.6 mm, and the fuel assembly pitch 215 mm. The effective core height is 3.66 m and the equivalent diameter 2.92 m. The core size corresponds to that of a current 3-loop plant which has a thermal power of some 2 700 MWt, i.e., the average power density of the JPSR core is about 75% of that in current PWRs. The average linear heat rate is relatively low, 13.2 kW/m.

For a fuel cycle length of 18 equivalent full power months, the U-235 enrichment of a refuelling batch for the equilibrium core amounts to 5.0 w/o. The calculated discharge burnup is as high as 58 000 MWd/t U. The refuelling and shuffling of fuel follow the out-in strategy using four batches; the four-batches refuelling strategy is adopted to obtain burnups of more than 45 GWd/t without increases of excess reactivity. The disadvantage of cycle length shortening is mitigated by the low core power density. The moderator density reactivity coefficients amount to 36 and 34% $\Delta k/k/g/cm^3$ at beginning and end of operating cycle, respectively, and the corresponding Doppler coefficients are -2.5×10^3 and -2.7×10^3% $\Delta k/k/^\circ C$.

The nuclear characteristics of the core have been evaluated by a three-dimensional core burn-up analysis with the SRAC nuclear calculation code system.

From the preliminary transient analysis, it was found that relatively large moderator density reactivity coefficients would be necessary to realize an inherent load following capability. In current PWRs, the upper concentration of soluble boron in hot, full power condition is limited so as to maintain a negative feedback at moderator density decrease.

In other words, the moderator density reactivity coefficient is too small to realize the desired inherent load following capability, especially at the beginning of a cycle.

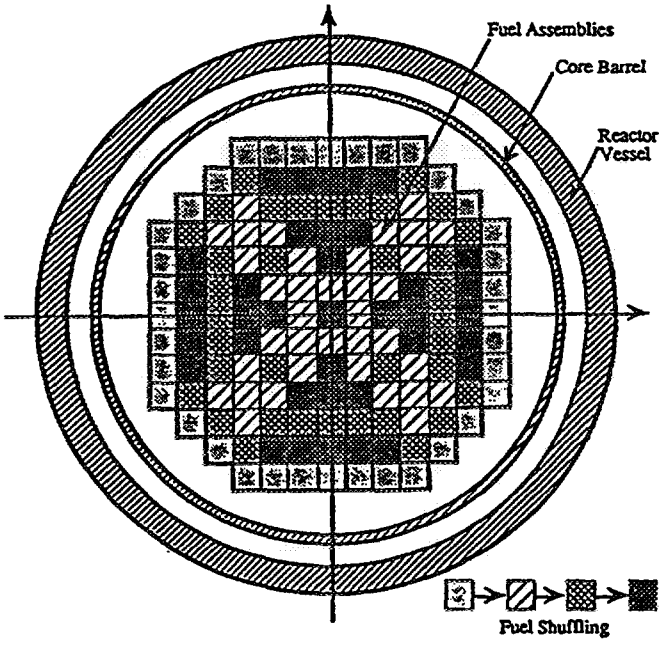


Figure 6.2-2 Horizontal cross-section of JPSR core

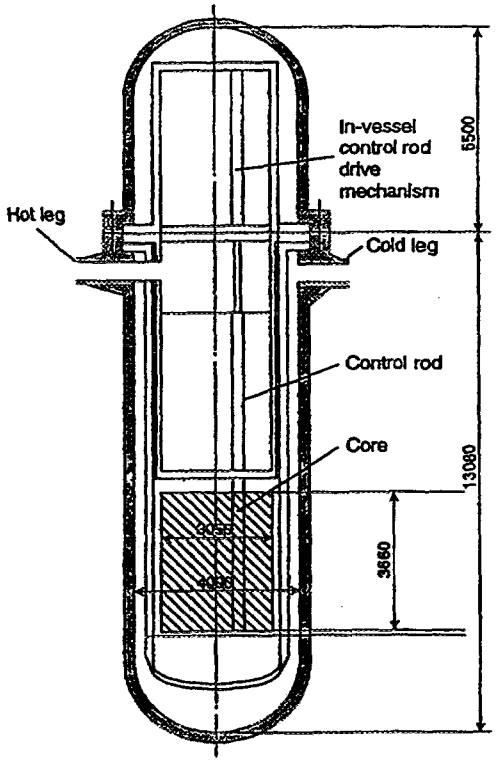


Figure 6.2-3 The JPSR reactor pressure vessel (600 MWe)

In JPSR, all excess reactivities are compensated by means of control rod movements. The reactivity coefficients of JPSR have been evaluated in the critical condition using control rods instead of soluble boron. The suitable reactivity characteristics expected from the transient analysis were obtained throughout the burnup periods.

The preliminary results on excess reactivity and control rod worth in cold condition at the beginning of cycle show that the total excess reactivity at cold conditions at the beginning of cycle amounts to 19.82% Δk/k (10.9 from burnup considerations, 1.85 from Xe, and 7.07 for power defect (HFP→CZP)). Several types of cluster control rods were considered and they were assumed to be inserted in all fuel assemblies in the core with the in-vessel control rod drive mechanisms. It was found that the conventional Ag-In-Cd control rods do not have enough reactivity worth (-23.58% Δk/k) to maintain a sufficient subcriticality (>5% Δk/k) in fuel exchanges. Utilization of B_4C control rods, - based on B_4C pellets with a 60% enrichment of B-10 yielding a control rod worth of -42.07% Δk/k, - seems to be feasible, but such a strong neutron absorber has the disadvantage of distorting the power distribution in operating condition. Therefore, it has been decided to use two different types of control rods: gray rods for controlling burnup reactivity changes and power distribution, and strong absorber rods for emergency shutdown and for holding subcriticality. Design details of the control rods are currently being optimized.

6.2.2.3 Fuel handling and transfer systems

Refuelling of the reactor is performed in the same way as for currently operating PWRs.

6.2.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel (Figure 6.2-3) has the same diameter as that for the AP-600 design concept of Westinghouse, but its height is larger than that for an ordinary three-loop PWR and for the AP-600 design (below the nozzle level) in order to incorporate in-vessel control rod drive mechanism (CRDMs). The diameter of the pressure vessel represents the only restriction with respect to raising the power, since the passive safety systems, without specific constraints can be adapted to a higher power level.

As noted above, the JPSR is designed to be operated without a chemical shim system. This means that the reactivity control capability of the soluble boron in the coolant, i.e., for compensation of the decreases in core reactivity due to the burnup of nuclear fuel and for ensuring an adequate reactivity margin at cold conditions, must be replaced by other means. To this end, the number of control rods and their rod drive mechanisms have been considerably increased. Installation of a sufficient number of ordinary control rod drive mechanism (CRDM) units will be very difficult, due to the technical restrictions in spacing of penetration holes on the top head of the pressure vessel for the connecting rods between the drive mechanisms and the control rods. Hence in-vessel CRDMs, which are being developed for a new marine reactor concept, the "MRX", at JAERI, have been adopted. The in-vessel CRDMs are, in addition to the reactivity control purposes, used also for active reactor shutdown and for damping of xenon oscillation, when necessary.

Elimination of the chemical shim and use of the in-vessel CRDMs represent a safety enhancement since this suppresses the probability of large reactivity insertion accidents due to boron dilution or control rod ejection.

Reactor internals

The reactor internals are similar to those of currently operating PWRs.

Steam generators

The JPSR is provided with two steam generators (SGs). In the once-through type steam generators the heat transfer is not determined by the temperature difference between the primary and secondary side coolants but governed by the feedwater flow rate. This provides for a very convenient way of controlling the reactor power output, simply by controlling the feedwater flow rate, - and thereby the heat removal rate of the steam generators, - with only small changes of the primary coolant temperature.

The quasi-steady state response characteristics of the primary loop on the changes in the heat removal rate have been estimated based on the assumed power density and reactivity coefficients of the reactor core. The thermal characteristics of the once-through steam generators of the JPSR were assumed to correspond to those of the Babcock & Wilcox (B&W) once-through type steam generators. The relation between the average coolant temperature and a normalized power was obtained from the balance between the reactivity coefficients of fuel temperature and moderator density. The dimensions of the JPSR steam generators were determined on the basis of the B&W design, so as to maintain steam temperatures within a range of less than 8K during quasi-steady power change from 0 to 100% of rated power. Actually, the tube length of the heated section is extended from 15.9 m of the B&W design to 19.3 m.

In a steam line break accident, or a steam generator tube failure including multiple tube failures, the main steam isolation valves will close automatically, and the steam generator shell side pressure will become close to the primary system pressure. The design pressure of the steam generator shell side can be the same as that of the primary coolant system and the main steam isolation valves should be directly connected to the shell of the steam generators. Radioactive material from the primary coolant can be contained in the shell side. In accident situations, heat removal through the steam generators is not expected in the JPSR.

Pressurizer

The expansion of the primary coolant was analyzed for a transient of 100% linear heat removal decay in 60 seconds. Based on the results for 50% linear heat removal decay, it was estimated that the maximum liquid expansion is 9.22 m^3 and a pressurizer with a steam volume of 60 m^3 can absorb the expansion with a pressure change from 15.6 to 19.83 MPa independent of pressure regulation. A pressurizer volume of 90 m^3 with a liquid volume of one third of the total volume is needed, about three times larger than that of currently operating PWRs of the same power level.

In order to reduce the pressurizer volume, the steam volume has been reduced to 30 m^3 and the total volume to 46.7 m^3. Therefore, pressure regulation is necessary for power transients that are larger than

50%. No pressure relief valve is necessary, however, since the liquid expansion can be absorbed in the large pressurizer for power transients from 100% down to 0% power.

Reactor coolant pumps

Canned motor pumps have been chosen for the main coolant pumps considering the high reliability of their performance and the associated simplification of the chemical and volume control system (CVCS). The main coolant pumps are located in the hot legs, since the coolant temperature variations at power level changes are smaller in the hot leg. In PWRs with once-through steam generators, the coolant lines are longer and the flow resistance is higher than in currently operating PWRs, and more pump power is needed. The rotating inertia of canned pumps is lower than that of the pumps in currently operating PWRs and their coastdown is faster; the transient behaviour of the reactor coolant loop must therefore be evaluated carefully.

Results from transient calculations show that a pump inertia that is 8% of the pump inertia for the pumps in currently operating PWRs, 3 110 kgm^2, can prevent the occurrence of DNB (departure from nucleate boiling) for the JPSR, even with failure of reactor scram, because of the core power reduction resulting from the density reactivity coefficient. Analyses with a subchannel code show that 6% is enough to meet the DNB criteria.

Main coolant lines

The JPSR has two main coolant loops, each consisting of one hot leg and two cold leg pipes. The hot leg is divided into two pipes at the main coolant pumps. The arrangement with two cold leg pipes selected for the JPSR has certain advantages; it reduces the break area size of a large cold leg break LOCA and therefore contributes to safety.

6.2.2.5 Reactor auxiliary systems

Chemical and volume control system (CVCS)

The adoption of a large pressurizer yields a dramatic simplification of the Chemical and volume control system (CVCS), since a volume control of the primary coolant is not needed during operation; only subsystems for primary coolant cleanup, chemical additives and hydrogen dissolving are necessary. The adoption of canned pumps as primary coolant circulation pumps eliminates the need for a seal-water supply subsystem. Charging and letdown systems are used for adjusting water level and system pressure in the pressurizer before nuclear heating in the startup period, but both systems are separated from the primary cooling system by valves during operation. The chemical (and volume) control system is installed in the containment, since the primary cooling system and its connecting components and piping are enclosed in the containment vessel so that radioactive material in the coolant can be retained in the containment in the event of pipe breaks. In practice, there may be some loss of coolant due to small leaks and sampling of water, but this has been taken into account in the margin of primary coolant inventory. The boron recycle system for removing boron from the coolant after events with borated water injection is operated only in reactor shutdown periods.

6.2.2.6 Operating characteristics

During load following operation, the reactor power will change automatically due to the changes in moderator density, and the ensuing reactivity changes, resulting from adjustments in the feedwater flow rate. Only the primary pressure control, using the pressurizer spray and heater systems, is needed. Such operation is realized by adoption of the boron free operation.

6.2.3 Description of turbine generator plant system

6.2.3.1 Turbine generator plant

Considering steam conditions such as 19.4K superheat, a reheater, regenerative type steam turbine was selected as the turbine generator. The turbine section consists of one high pressure and two low pressure turbines. Two stages of moisture separator reheater and six stages of steam extraction are adopted.

6.2.3.2 Condensate and feedwater systems

Two units of condenser is adopted. The radial flow type condenser with horizontal tube bundle is adopted as the current PWRs. The cooling water is sea water. The temperature of cooling water at the inlet of the heat exchanger was assumed to be 19°C as the design condition. The heat transfer area of the tubes was determined to be 47 000 m^2 for a heat transfer rate of 1.04×109 kcal/h assuming an inlet temperature of 26°C considering the summer condition around the JAERI site. There are three three condensate pumps, each with a capacity of 50% of the total flow rate of 1300 m^3/h; i.e, during normal operating conditions two of the pumps are operated and one is at standby. The pumps are motor operated, vertical shaft-multistage type pumps with a rated head of 70 m. The condensed water is pumped by the condensate pumps through a ground steam condenser, two of three booster pumps (one is at standby), two parallel placed feedwater heaters (each has four stage heaters) and a deaerator, and then enters the feedwater system.

The feedwater flows through two of three feedwater booster pumps, two of three main feedwater pumps and two high pressure feedwater heaters placed in parallel and is supplied to the steam generators. The feedwater booster pump is a motor operated, vertical shaft-multi-stage type pump. The main feedwater pumps are motor operated, vertical shaft, centrifugal type pumps, each with a capacity of 50% of the rated flow rate of 2300 m^3/h and with a rated head of 655 m. Two high pressure feedwater heaters in parallel are adopted. The heaters are of horizontally placed shell and tube type with a heat transfer area of 1600 m^2.

6.2.3.3 Auxiliary systems

The sea water system is needed to remove the heat from the condansate system. This system consists of two sea water feed pumps and two lubrication water pumps (one is for standby). The sea water feed pumps are vertical shaft, movable vane type pumps. The flow rate of the feeding sea water can be changed by moving the vanes. Each pump has a rated flow rate of 50% of the total flow rate and a rated head of 13 m. The feed pumps need lubrication of the shaft and for that there are two lubricating water pumps with a capacity of 20 m^3/h.

6.2.4 Instrumentation and control systems

6.2.4.1 Design concepts including control room

No information provided.

6.2.4.2 Reactor protection system and other safety systems

In case of unusual conditions, the reactor protection system takes over and automatically scrams the reactor and actuates the relevant systems. However, most of the safety systems actuate automatically due to adoption of passive concepts.

6.2.5 Electrical systems

6.2.5.1 Operational power supply systems

No information provided.

6.2.5.2 Safety-related systems

Since JPSR adopts passive systems for coolant injection in LOCA conditions, an actuation of a pump for coolant injection is not needed. Furthermore, the JPSR does not need a spray system for suppression of the containment pressure. Therefore, under emergency conditons, required electrical power supply in JPSR is less than in currently operating PWRs. Consequently, the emergency diesel generator is of small capacity and is considered as a non-safety grade component; this contributes to cost reduction.

6.2.6 Safety concept

6.2.6.1 Safety requirements and design philosophy

The JPSR concept aims at attaining an enhanced safety and a reduction of the manpower needed for operation and maintenance, and JAERI considers that these objectives can achieved through simplification of plant systems by utilization of passive safety systems and by enhancement of the inherent safety characteristics of the reactor core. The adoption of passive systems and inherent safety features simplifies the reactor system and makes the operation easier; hence, it results in reduction of manpower needs and also reduces the effects of human errors on safety.

The inherent safety characteristics of the reactor core, which makes the core power automatically matching a change in the heat removal rate, are achieved by an increased reactivity coefficient feedback from moderator density changes, a result of eliminating the soluble boron from the coolant. This contributes to an enhanced safety, since the core power will decrease more rapidly during accidents with degradation of the heat removal capability such as loss-of-load and failure of the reactor scram than in currently operating PWR plants using chemical shim operation. A further advantage of the strong feedback is that the core power can follow rapid changes in the heat removal rate of the steam generator without need for operation of complex controls, such as movement of the control rods.

The passive safety systems consist of the passive residual heat removal (RHR) system, with upper and lower RHRs, and the passive containment cooling system for removing the heat generated in the core to the atmosphere, and the advanced accumulator and the gravity coolant injection systems for coolant injection into the primary system under accident conditions. The accidents are considered to be divided into two groups: loss-of-coolant accidents (LOCAs) and non-LOCA accidents. The safety in non-LOCA accidents, in particular in the event of accidents involving degradation of heat removal rate, is ensured by means of the upper RHR which is a completely passive system.

This system which is described in the next section, contributes significantly to a reduction of the CDF (core damage frequency). It has been analyzed, by calculations performed with the thermo-hydraulic computer code RETRAN-02, that the upper RHR can maintain the core cooling capability in the event of a hypothetical complete loss-of-heat-sink accident, taken as the most challenging condition for evaluation of the adequacy of the design. The calculated results (Figure 6.2-4) show that the upper RHR functions properly and transfers the heat to the gravity driven coolant injection pool.

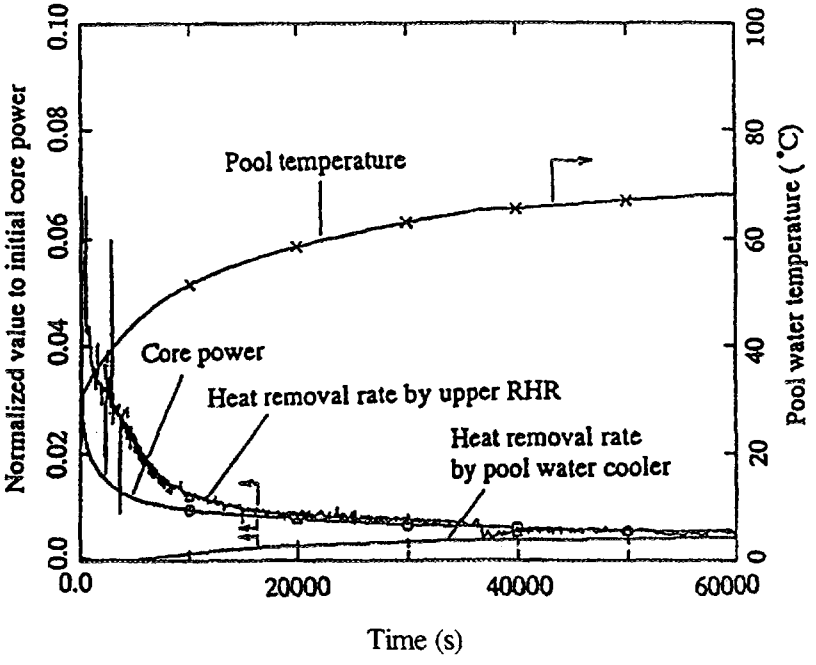


Figure 6.2-4 JPSR - Analysis of loss of heat sink

The water of this pool heats up from some 30°C to about 70°C during the first 16 hours, and the heat removal rate of the pool water cooling system increases, with the increasing temperature, to about 0.4% of the initial core power during this period.

In the event of LOCAs, the long term cooling capability is ensured by the lower RHRs. The lower RHRs also serve to reduce the primary system pressure leading to early actuation of the accumulator system and the gravity coolant injection system. The need for a supplementary, active depressurization system using depressurization valves to reduce the primary system pressure is currently under consideration. Injection of coolant is achieved by means of the accumulator system and the gravity coolant injection system. The coolant in the gravity coolant injection pool is also injected into the reactor cavity via the gravity coolant injection system and break portions can be submerged by the injected coolant.

Transient thermal-hydraulic analyses of cold leg break LOCAs performed with the RETRAN-02 code have showed that the present system could ensure the safety for a wide range of break areas. For a small break cold leg LOCA with 5% break area, the results showed quite clearly that the primary system pressure remains high without the passive RHR, whereas it decreases to the actuation level of the accumulators in about 200 seconds by means of the lower RHR. In a large break LOCA, the results showed that the maximum cladding temperature during the blowdown phase remains low enough, due to large mass inventory inside the RPV.

On the other hand, nuclear technologies currently used are also adopted as much as possible, in order to minimize developmental work to make the JPSR feasible.

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

The JPSR adopts many passive safety systems for basic safety functions, e.g., for residual heat removal and for maintaining the coolant inventory. The most important systems are: the passive residual heat removal system (PRHR); the reactor coolant inventory control (RCIC); and the passive containment cooling system (PCC).

Compared with the safety systems of currently operating PWRs, active safety components such as high and low pressure injection systems, auxiliary feedwater system and containment spray system are excluded in the JPSR safety systems. Also, the ultimate heat sink of JPSR is the atmosphere via a natural circulation air cooling system and does not rely on the component cooling system and sea water system as in current PWRs.

Passive residual heat removal (PRHR) system

The passive residual heat removal system consists of RHR heat exchangers, a gravity-driven coolant injection pool and air coolers. Two lower and one upper RHR heat exchanger are installed outside the pressure vessel.

The lower RHR heat exchangers are connected to the primary hot legs and cold legs with two types of valves in parallel. The lower RHR loops are activated by opening of one of these valves. The lower RHR loops are used for heat removal during every transient and accident conditions including loss-of-coolant accidents (LOCAs).

The upper RHR heat exchanger is installed between the pressurizer and the hot leg which is not connected to the pressurizer. The upper RHR loop has no valve and the natural circulation cooling is actuated automatically when the water level in the pressurizer rises due to an increase in primary water temperature.

The secondary sides of both the upper and lower RHR heat exchangers are connected to the gravity-driven coolant injection pool and cooled by natural circulation of water at atmospheric pressure. Each of the passive RHR loops is designed to remove at least 6% of full power, which corresponds to the core power just after reactor shutdown, by natural circulation even when the pool temperature is as high as 100°C. In

order to ensure natural circulation cooling in the case of LOCAs, the lower RHR heat exchangers are submerged.

Reactor coolant inventory control system

The passive engineered safety injection system of improved design consists of advanced accumulators and gravity-driven coolant injection systems. Each system comprises two units for redundancy.

The advanced accumulator is a flow-controlled accumulator developed by Mitsubishi Heavy Industries, Ltd. (MHI). When the system pressure decreases to a certain level, the accumulators are actuated by check valves opening and water flows into the primary system under nitrogen gas pressure. The flow rate is initially high and then reduced to about 20% of the initial flow rate by passive means. The gravity-driven injection lines are connected to the cold legs and the reactor cavity for primary loop injection and cavity flooding, respectively.

Passive containment cooling system

The gravity-driven coolant injection pool is located inside the containment and its water is cooled by air cooler units placed outside the containment. The core residual heat is removed by natural circulation through the primary loop of the RHR system and then transferred to the pool by natural circulation through the secondary loop of the RHR system. The pool is in its turn cooled by natural circulation of water at atmospheric pressure through the pool water heat exchangers and the heat is finally transferred to the atmosphere through the air cooler units. Six units of natural circulation cooling systems are installed, out of which four will provide sufficient heat removal capacity to remove 1% of full power. The imbalance between the heat removed by the RHR system and the heat transferred to the atmosphere by the air cooler units is absorbed by the water mass in the pool, increasing its temperature.

The primary coolant system is contained inside an isolation boundary which is connected to the gravity-driven coolant injection pool by vent pipes. In the case of LOCA events, the primary coolant is discharged into the pool water through the vent pipes and steam is condensed in the pool. Therefore, a containment spray system is not necessary because the pool can be used for the absorber of decay heat and radioactive iodine.

Long-term cooling

The long-term cooling is accomplished by the passive heat removal systems, i.e., the upper and lower RHRs and the containment cooling system.

6.2.6.3 Severe accidents (Beyond design basis accidents)

No information provided.

6.2.7 Plant layout

6.2.7.1 Buildings and structures, including plot plan

No plot plan provided.

Seismic behaviour and Design requirements

No information provided.

6.2.7.2 Reactor building

The reactor building is being designed at present.

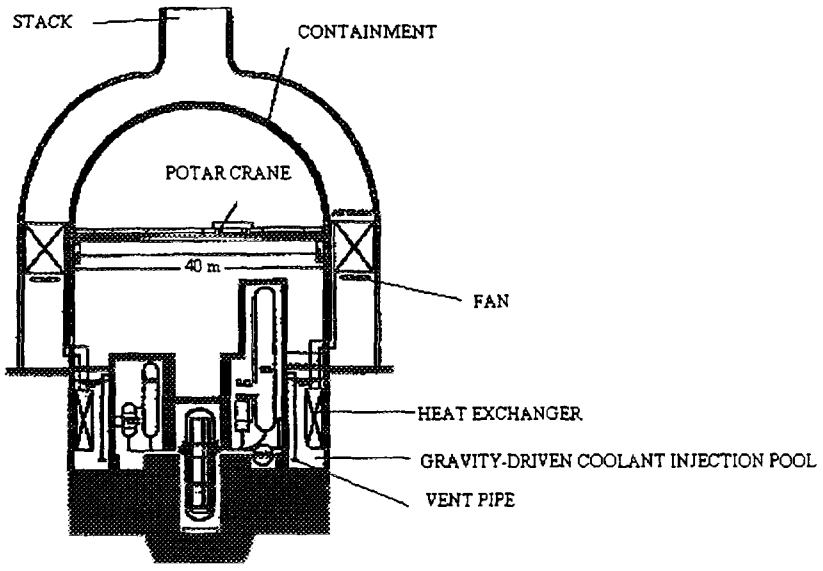


Figure 6.2-5 The containment concept of the JPSR

6.2.7.3 Containment

The primary containment (Figure 6.2-5) is a free standing, cylindrical steel structure with a hemispherical top, sunk several tens of meters into the ground. It has an internal diameter of 40 m and a total height of 78.5 m.

The containment bottom encloses a massive concrete structure which at its centre contains a rather narrow reactor vessel cavity with some sort of insulation at its bottom. The upper portion of this structure contains compartments for the major components of the reactor system and for the two sections of the gravity driven coolant injection pool (Figure 6.2-6).

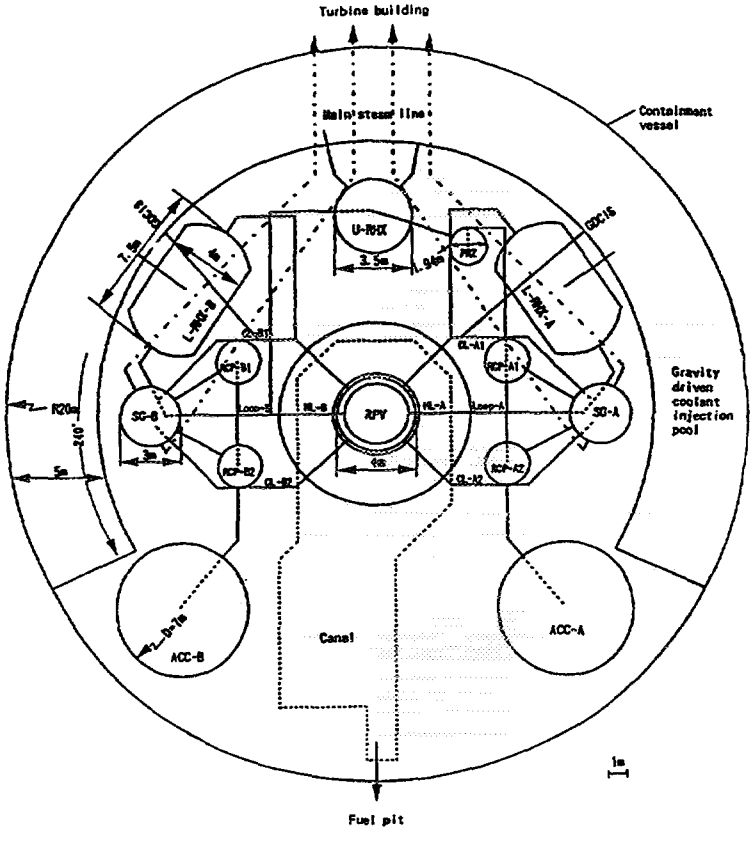


Figure 6.2-6 Layout of major components in JPSR containment

The component compartments are separated from the pool sections and from the upper portion of the containment by means of leaktight walls and roofs; the reactor shaft is closed with a removable cover. The separated compartments are connected to the gravity driven coolant injection pool sections by means of vent pipes which will convey steam released at a rupture of the reactor coolant pressure boundary into the pool water where it condenses. In other words, these portions of the containment has a pressure suppression function.

The containment shell is surrounded by an external structure with a diameter of 60 m, serving as a flow guiding structure for cooling air. Air inlets are provided at ground level, and an air outlet, a stack, is arranged at the top. The air coolers of the passive containment cooling system are installed in the annulus between the containment shell and the surrounding structure. During normal operation, fans are providing circulation of ventilation air up through the coolers and the annulus, and cooling of the gravity driven coolant injection pools inside the containment.

6.2.7.4 Turbine building

The layout of the turbine generator system in the building is shown in Figure 6.2-7. The turbine building is assumed to be located at the south side of the reactor building, with the sea side to the south side of the building. The layout of the components is determined so as to save building space while at the same time ensuring enough space for maintenance activities. The size has been determined to 92 m × 42.5 m x 40.7 m (W × D × H).

The lowest basement floor is at 14.5 m below the ground level. In order to save the space, the first and second feedwater heaters are placed inside the condenser dome. The third and fourth feedwater heaters are placed on the piping line connecting to the deaerator.

6.2.7.5 Other buildings

No information provided.

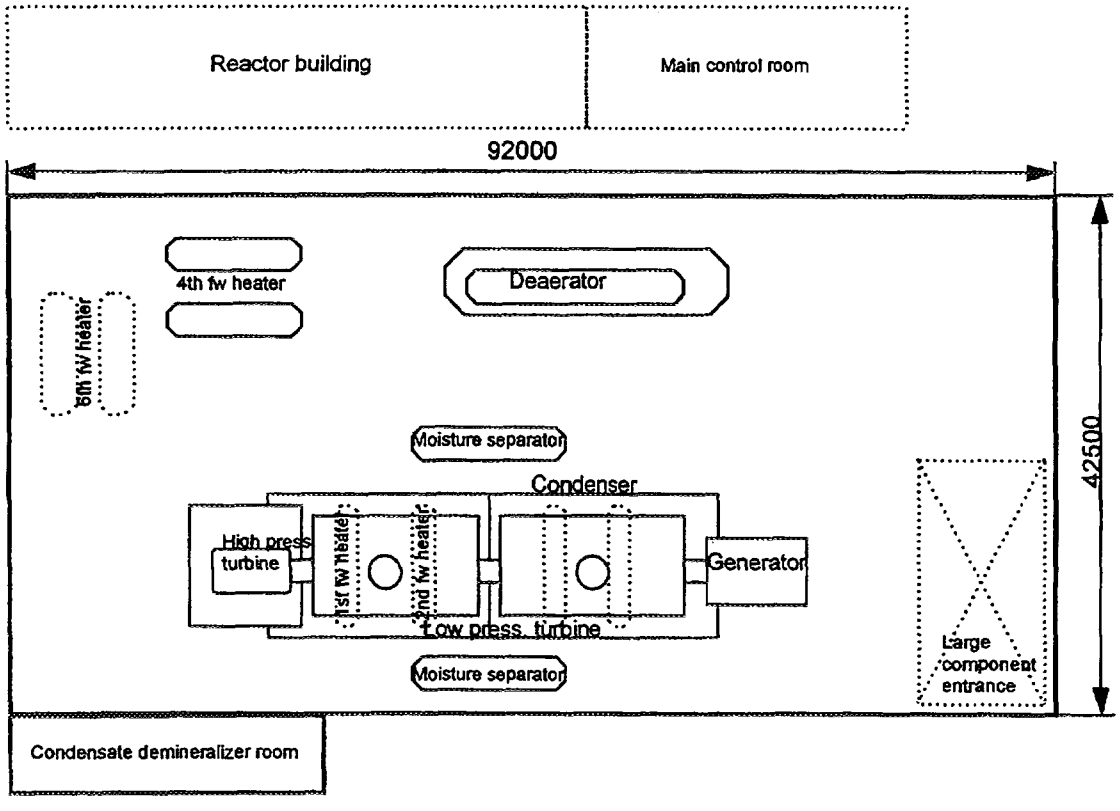


Figure 6.2-7 General layout of turbine building (view from above)

6.2.8 Technical data

General plant data

Power plant output, gross	659	MWe
Power plant output, net	630	MWe
Reactor thermal output	1 853	MWt
Power plant efficiency, net	34	%
Cooling water temperature	19	°C

Nuclear steam supply system

Number of coolant loops	2 hot legs/4 cold legs	
Primary circuit volume, including pressuriser	366	m3
Steam flow rate at nominal conditions	1 031	kg/s
Feedwater flow rate at nominal conditions	1 031	kg/s
Steam temperature/pressure	299/6.35	°C/MPa
Feedwater temperature/pressure	243/	°C/MPa

Reactor coolant system

Primary coolant flow rate	8 888	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	285	°C
Coolant outlet temperature, at RPV outlet	325	°C
Mean temperature rise across core	40	°C

Reactor core

Active core height	3.66	m
Equivalent core diameter	2.92	m
Heat transfer surface in the core	4 181	m^2
Fuel inventory	67	t U
Average linear heat rate	13.2	kW/m
Average fuel power density	27.7	kW/kg U
Average core power density (volumetric)	75.6	kW/l
Thermal heat flux, Fq	443.2	kW/m^2
Enthalpy rise, F_H	2.097	

Fuel material	Sintered UO_2	
Fuel assembly total length	4 058	mm
Rod array	square, 17×17	
Number of fuel assemblies	145	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	25	
Number of spacers	9	
Enrichment (range) of first core		Wt%
Enrichment of reload fuel at equilibrium core	5.0	Wt%
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel	58 000	MWd/t
Cladding tube material	Zircaloy	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly	640	kg
Active length of fuel rods	3 660	mm
Burnable absorber, strategy/material		
Number of control rods		
Absorber rods per control assembly		
Absorber material		
Drive mechanism	in-vessel type	
Positioning rate		
Soluble neutron absorber		

Reactor pressure vessel

Cylindrical shell inner diameter		4 000	mm
Wall thickness of cylindrical shell		200	mm
Total height		18 580	mm
Base material:	cylindrical shell	JIS SFVQ 1A	
	RPV head	JIS SFVQ 1A	
	liner		
Design pressure/temperature		17.25/343	MPa/°C
Transport weight (lower part), including head		513	t
	RPV head	140	t

Steam generators

Type	Vertical, Once-through	
Number [Thermal capacity 970 MWt]	2	
Heat transfer surface (per SG)	10 016	m^2
Number of heat exchanger tubes	10 389	
Tube dimensions	15.9/14.18	mm
Maximum inner diameter	3 000	mm
Total height	24 320	mm
Transport weight	365	
Shell and tube sheet material	JIS SFVQ 1A	
Tube material	Inconel-600	

Reactor coolant pump

Type	Single-stage, centrifugal pump with canned motor	
Number	4	
Design pressure/temperature	17.25/360	MPa/°C
Design flow rate (at operating conditions)	2 222	kg/s
Pump head	110	m Wg
Power demand at coupling, cold/hot	-/	kW
Pump casing material	Stainless steel	
Pump speed	1906	rpm

Pressuriser

Total volume	46.7	m^3
Steam volume: full power/zero power	30.0/	m^3
Design pressure/temperature	17.25/360	MPa/°C
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter	1 940	mm
Total height	15 740	mm
Material	JIS SFVQ 1A	
Transport weight	243	t

Pressuriser relief tank

	Not applicable	
Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Dry, free standing in steel with pressure-suppression	
Overall form (spherical/cyl.)	cylindrical with spherical head	
Dimensions (diameter/height)	40/76.2	m
Free volume	41 400	m^3
Design pressure/temperature (DBEs)		MPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate		vol%/day
Is secondary containment provided?	No	

Reactor auxiliary systems

Reactor water cleanup, capacity		kg/s
filter type		
Residual heat removal, at high pressure		kg/s
at low pressure		kg/s
Coolant injection, at high pressure		kg/s
at low pressure		kg/s

Power supply systems

Main transformer, rated voltage		kV
rated capacity		MVA
Plant transformers, rated voltage		kV
rated capacity		MVA
Start-up transformer rated voltage		kV
rated capacity		MVA
Medium voltage busbars (6 kV or 10 kV)		
Number of low voltage busbar systems		
Standby diesel generating units: number		
rated power		MW
Number of diesel-backed busbar systems		
Voltage level of these		V ac
Number of DC distributions		
Voltage level of these		V dc

Number of battery-backed busbar systems		
Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	Reheater, regenerative Steam turbine	
Number of turbine sections per unit (e.g. HP/LP/LP)	1HP/2 LP	
Turbine speed	1500	rpm
Overall length of turbine unit	34.5	m
Overall width of turbine unit	9.7	m
HP inlet pressure/temperature	6.09/295.7	MPa/°C

Generator

Type		
Rated power		MVA
Active power		MW
Voltage		kV
Frequency		Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type	Radial flow type	
Number of tubes		
Heat transfer area	47 000	m^2
Cooling water flow rate	42.875	m^3/s
Cooling water temperature	19	°C
Condenser pressure	963	hPa

Condensate pumps

Number	3 × 50%	
Flow rate (each)	361	kg/s
Pump head	70	m
Temperature	33.1	°C
Pump speed		rpm

Condensate clean-up system

Full flow/part flow	
Filter type	

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/ °C

Feedwater pumps

Number	3 × 50%	
Flow rate (each)	638.9	kg/s
Pump head	655	m
Feedwater temperature	230.8	°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages	6 (1 HP, 5 LP)
Redundancies	2

6.2.9 Project status and planned schedule

The JPSR design project was initiated at JAERI in the fiscal year 1993. The first stage of the design development which aims at establishing the main features of the design, was scheduled to be completed in three years.

The present design concept is now almost determined regarding the primary system and the passive components. Design work on the other parts of the plant is under way.

Design improvements of the concept will continue.

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6.3 PIUS, ABB ATOM, SWEDEN

6.3.1 Introduction

PIUS was originally an acronym that referred to a design philosophy for a new series of reactor designs in which all functions and measures needed to protect the reactor core and prevent overheating and degradation of its fuel (the ultimate safety) should be inherent in the process, without need for initiation or actuation from operators or control systems and dependence of power supply. Eventually, it became the designation of a new light water cooled power reactor design concept that is being developed by ABB Atom in Sweden.

A furthered development of the established light water reactor technology to make it better adapted to future needs is seen as a prerequisite for nuclear power to remain a viable option for generation of electrical energy in the coming decades, in industrialised as well as developing countries. Simplifications, improved economy and a "good neighbour" image are considered areas of particular importance, and the development of the PIUS concept follows that track; it represents an effort to accomplish a simplified reactor design which can be more easily understood by the general public.

The design objectives that were established in the 1970s by ABB Atom at the start of the development work on the first of the "PIUS" reactor types, the heat-only reactor, SECURE, encompassed:

- It should be competitive in small and moderate capacity units with respect to costs, availability and maintainability;
- It should be based on demonstrated widely employed basic technology to a maximum extent;
- It should be simple and flexible to operate and not make excessive demands on the resources of qualified personnel;
- The safety should be "transparent", i.e., understandable to educated laymen, built on simple natural laws, and independent of failure-prone systems and components;
- It should be operator forgiving, i.e., the "human factor" as a risk element should be largely eliminated by design;
- It should, from a technical point of view, be safe enough to be located almost anywhere, even in densely populated areas;
- It should be capable of surviving extreme external conditions without risk of environmental radioactive contamination.

These design objectives were carried over to the work on PIUS for power production. PIUS is basically a pressurized water reactor (PWR) in which the primary system has been rearranged in order to accomplish an efficient protection of the reactor core and the nuclear fuel by means of thermal-hydraulic characteristics, in combination with inherent and passive features, without reliance on operator intervention or proper functioning of any mechanical or electrical equipment. Together with wide operating margins, this should make the plant design and its function, in normal operation as well as in transient and accident situations, much more easily understood and with less requirements on the capabilities and qualifications of the operators.

6.3.2 Description of the nuclear systems

6.3.2.1 Primary circuit and its main characteristics

PIUS is a new reactor concept based on well established LWR technology and infrastructure. It is a passive and simplified, reconfigured PWR incorporating also some BWR features, with a nominal power output of 600 MWe.

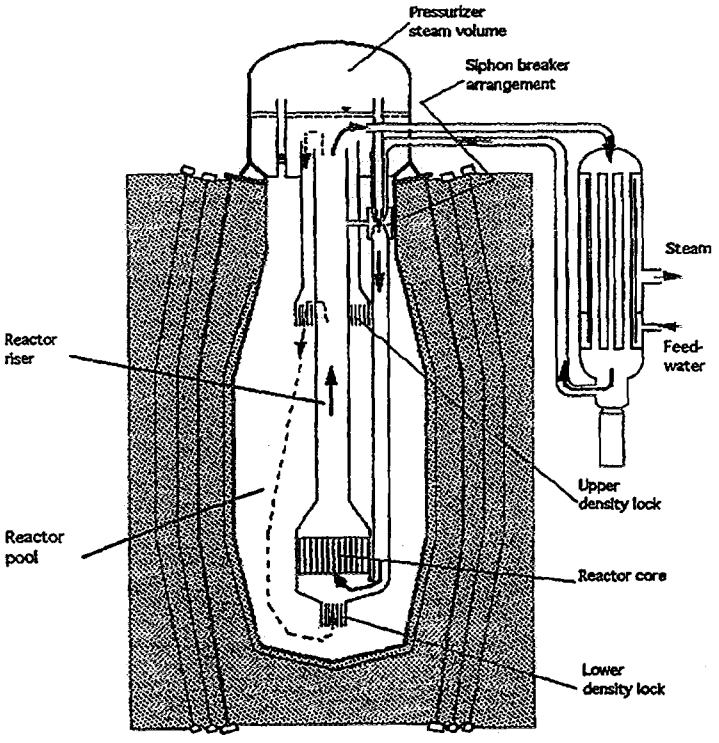


Figure 6.3-1 PIUS - Principle arrangement

The arrangement is somewhat modified reflecting the design goals of achieving increased simplicity and safety, in particular with respect to protection of the reactor core in possible accident scenarios. The basic arrangement principle of PIUS is outlined in Figure 6.3-1, and a main flow diagram for the power plant is presented in Figure 6.3-2.

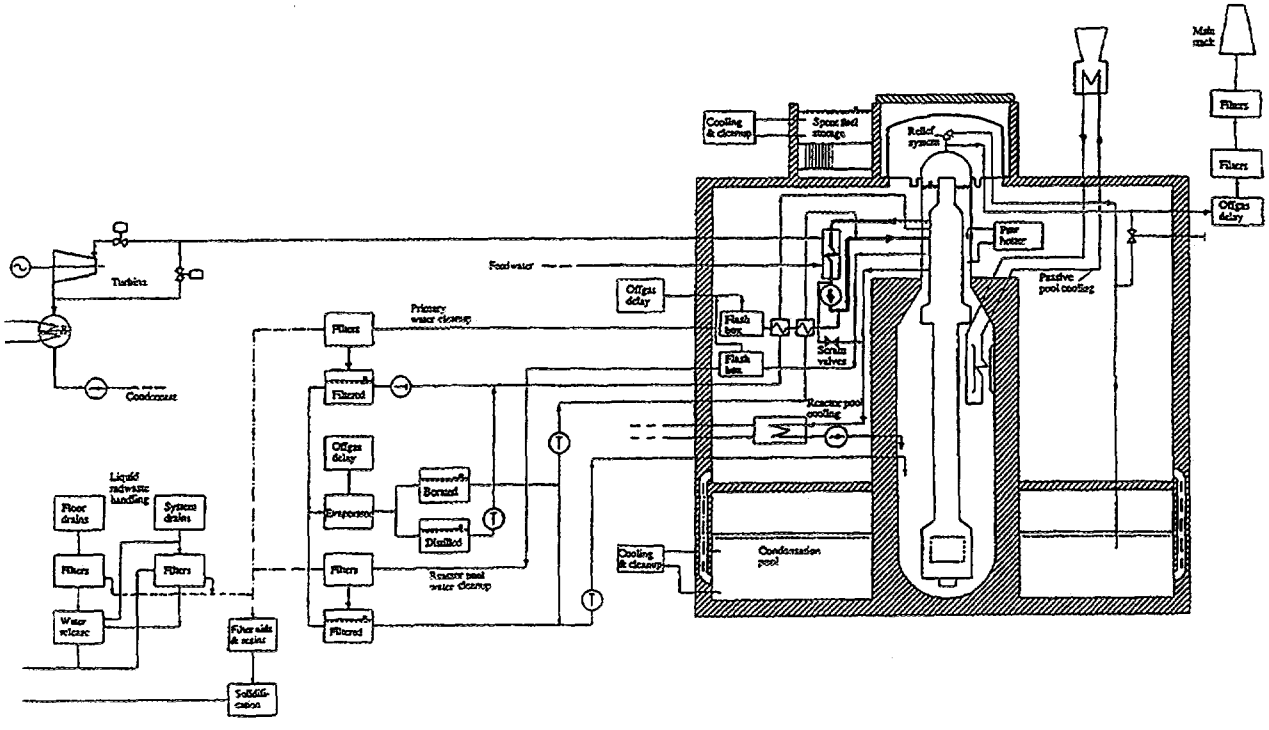


Figure 6.3-2 PIUS - Main flow diagram

The reactor core is an open PWR type core located near the bottom of the reactor pool, which is a high-boron content water mass enclosed by a prestressed concrete vessel. The PIUS reactor does not use control rods, neither for reactor shutdown nor for power shaping. Reactivity control is accomplished by means of reactor coolant boron concentration control (chemical shim) and by coolant (moderator) temperature control.

From the core, the heated coolant - at a temperature of 290°C - passes up through the riser pipe, and leaves the reactor vessel through nozzles in its upper steel part. The coolant continues in hot leg coolant pipes to four straight tube once-through steam generators. The main coolant pumps are located below the steam generators, and structurally integrated with these. The pumps are sized-up versions of the glandless, wet motor design pumps that have been utilized as recirculation pumps in the ABB Atom BWR plants.

The cold leg piping enters the reactor vessel at the same level as the hot leg nozzles, and the 260°C return flow is directed downwards to the reactor core inlet via the downcomer.

On its way down, the flow velocity is increased in a siphon breaker arrangement with open connections to the pressurizer. The siphon breaker is intended to prevent siphoning off too much reactor pool water inventory in the hypothetical event of a cold leg rupture. During normal operation, the siphon breaker does not affect the water circulation. At the bottom of the annular downcomer the return flow enters the reactor core inlet plenum.

Below the core inlet plenum there is a pipe opening (of less than one meter diameter) towards the enclosing reactor pool. Inside this pipe there is a tube bundle arrangement to minimize water turbulence and mixing and to ensure stable layering of hot reactor primary loop water on top of colder reactor pool water. This pipe with the bundle arrangement and the stratified water is called the lower "density lock". The position of the interface between hot and cold water is determined by temperature measurements, and this information is used for controlling the speed (or flow rate) of the main coolant pumps to maintain the interface level at a constant position during normal operation. The upper portion of the density lock pipe is normally filled with hot primary loop water, serving as a buffer volume to prevent ingress of pool water and spurious reactor shutdowns at minor operational disturbances.

There is another "density lock" arrangement at a high location in the pool, connected to the upper riser plenum - the volume on top of the riser from which the water is drawn into the hot leg pipes. This upper density lock has a similar arrangement of tube bundles and a buffer volume above the hot/cold water interface level. There are also a number of small openings between the riser and the density lock.

This reactor system configuration - with the two always open density locks - is the basis for an exceptional safety performance of PIUS plants. There is always an open natural circulation path through the core, going from the reactor pool to the lower density lock, to the core via inlet pipes, through the core itself, the riser, the passage from the upper riser plenum (and through the direct riser - density lock connections), and the upper density lock back to the pool. During normal plant operation, the natural circulation circuit is kept inactive by means of the speed control of the main coolant pumps, maintaining the hot/cold interface in the lower density lock at a constant position.

In a PIUS plant, the core coolant flow rate is determined by the temperature at the reactor core outlet - relative to the reactor pool. The resulting pressure drop across the core and up through the riser must correspond to the static pressure difference between the interface levels in the upper and lower density locks. The main coolant pumps are operated to establish a pressure balance across the lower density lock to keep the reactor system in operation. In case of a severe transient or an accident, the natural circulation flow loop will be established, providing both reactor shutdown and continued core cooling.

The hot/cold interface level in the upper density lock is determined by the total volume of the primary loop water mass, when the position of the interface level in the lower density lock is kept constant. The interface level in the upper lock is, in the same way as in the lower density lock, monitored by temperature measurements, and its position can be controlled by adjusting the volumes (or masses) of the reactor pool

and primary loop. The temperature measurements in the density lock are basically used for direct control of the reactor pool volume, whereas the reactor primary loop volume is controlled by level measurements in the pressurizer. The hot parts of the primary system are isolated from the cold reactor pool water by means of a wet thermal insulation of metallic type. This insulation consists of a number of parallel, thin stainless steel sheets with stagnant water between them.

The water in the reactor pool is cooled by two systems; one with forced circulation of pool water through out-of-vessel heat exchangers and pumps, and one entirely passive system utilizing coolers submerged in the reactor pool and natural cooling water circulation loops up to dry natural draft cooling towers located on the top of the reactor building.

The natural cooling water circulation system ensures the cooling of the reactor pool in accident and station blackout situations, and prevents boiling of the reactor pool water inventory. In the hypothetical case that all pool cooling systems fail, the water inventory ensures the core cooling for a protracted period of time (7 days).

6.3.2.2 Reactor core and fuel design

The reactor core is an open PWR type core made up of 213 fuel assemblies with standard PWR fuel rod diameter and a reduced height. The 2000 MWt core is located near the bottom of the reactor pool, which is a high-boron content water mass enclosed by a prestressed concrete vessel. The PIUS reactor does not use control rods, neither for reactor shutdown nor for power shaping. Reactivity control is accomplished by means of reactor coolant boron concentration control (chemical shim) and by coolant (moderator) temperature control.

The core data are significantly relaxed in comparison with current PWR practice in terms of average linear heat load, temperatures, flow rates and associated pressure drops. Power shaping at the beginning of an operating cycle and reactivity compensation for burnup are accomplished by means of a burnable absorber (gadolinium) in some of the fuel rods. This means that the boron concentration can be kept at a rather low level, throughout the operating cycle, and the moderator temperature reactivity coefficient will be strongly negative under all operating conditions.

6.3.2.3 Fuel handling and transfer systems

During refuelling operations the containment dome and the reactor vessel head are removed, and the cavity above the dome is filled with water. The reactor internals are lifted out in sections, and placed in the water-filled cavity. The refuelling is carried out with a conventional refuelling machine from the reactor service room. Fresh fuel is brought into the cavity from a fresh fuel storage in the reactor building, and the spent fuel is removed to an adjacent spent fuel pool at the reactor service room floor level.

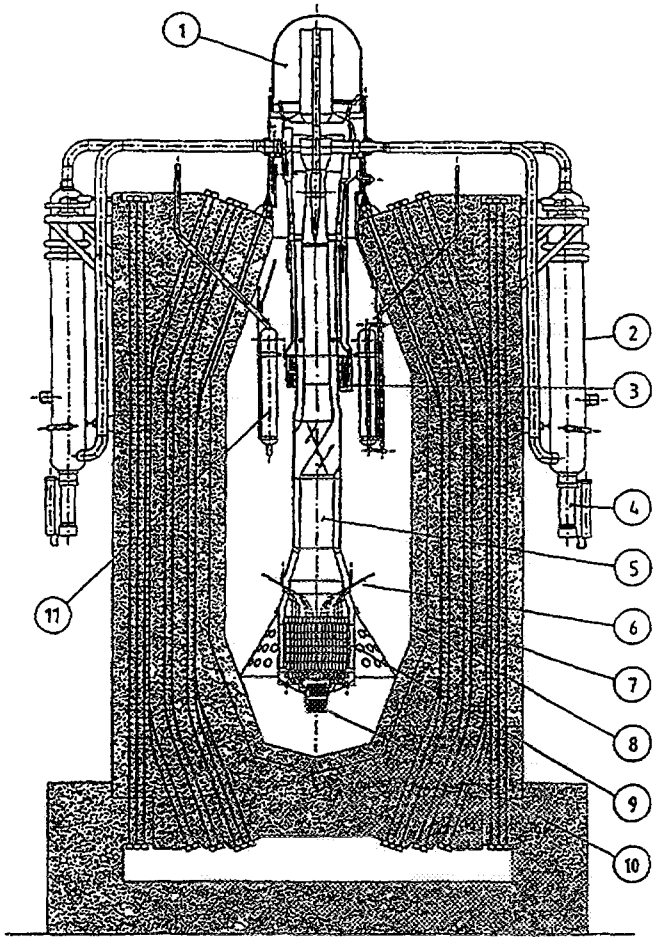
6.3.2.4 Primary components

The principal features of the nuclear steam supply system (NSSS) of PIUS are delineated on Figure 6.3-3.

Reactor pressure vessel

The reactor pressure vessel is made up of two parts: a prestressed concrete reactor vessel (PCRV) and a steel pressure vessel upper part (PVUP).

The prestressed concrete vessel has a cavity with a diameter of about 12 and a depth of about 38 m, containing some 3 300 m^3 of water. The concrete vessel is a monolith with a cross-section of about 27 and a height of about 43 m. It is anchored to the foundation mat structure by means of prestressing tendons. The pressure retaining capability of the vessel is ensured by a large number of prestressing tendons - partly horizontal tendons run around the cavity, partly vertical tendons run from the top to the bottom, - and by reinforcement bars.



1. Pressurizer steam volume
2. Steam generator (4)
3. Upper density lock
4. Main coolant pump (4)
5. Riser
6. Core instrumentation
7. Embedded steel membrane
8. Pool liner
9. Core
10. Lower density lock
11. Submerged pool cooler, cooled in natural circulation by ambient air.

Figure 6.3-3 PIUS - Principal features of the NSSS

The inside of the cavity is provided with a stainless steel liner. In addition, there is a second barrier - an embedded steel membrane about 1 m into the concrete - up to a level above the upper density lock to ensure that the reactor pool water volume below this level can not be lost by liner leakage. Concrete vessel penetrations are not permitted below this level.

On top of the prestressed concrete vessel there is a steel vessel extension which is fixed by means of separate tendons anchored to the bottom of the concrete vessel. This extension contains the pipe nozzles for the hot and cold leg pipes, for the forced circulation loops of the reactor pool cooling system, and for some other system pipes. It also encloses the upper riser plenum, and the pressurizer.

The four steam generators are located on two sides of the concrete vessel. The two other sides are utilized for installation of equipment associated with supporting systems, the containment HVAC systems, etc.

Reactor internals

The reactor internals comprise the reactor assembly and its ancillary equipment inside the concrete vessel cavity and the reactor vessel upper parts. The reactor assembly mainly consists of the thin-walled flow guide structures that direct the coolant to and from the reactor core and the core supporting structures, but it does not include the main pressure retaining vessel.

The reactor assembly is supported by a support skirt - a truncated cone that at its lower end is welded to the cavity liner in the lower conical portion of the concrete vessel cavity. The cone is penetrated by a large number of holes through which pool water may flow, i.g., when the pool loop is activated. A seismic restraint can be arranged for the reactor assembly at a higher location if the plant site is in a high seismicity area.

The internals of the assembly consist of core support components, core water supply components and the lower density lock. The core envelope structure encloses the core and the core support components as well as the core water supply components. It is a partly double-walled cylindrical vessel of which the inner wall serves the flow guiding function of the core barrel in a present day PWR. The core bottom and side support structures are made up of a bottom support plate and a core baffle, in a similar way as in other PWRs. The core water supply components have a two-fold task corresponding to the two loop core cooling arrangement of PIUS; during normal operation, they distribute the recirculated coolant from the steam generators to the fuel assemblies and deliver it to the riser, and when the reactor pool loop is activated, they deliver borated pool water to each fuel assembly. The riser which serves to pull coolant through the reactor core and deliver it to the hot legs, and the downcomer which conveys the coolant back from the cold legs to the core inlet, are arranged concentrically in four flow guide structures. The two lowest structures rest on the core support structures whereas the two other are hanging from a flange in the pressure vessel upper part. The two halves are kept together by a flexible bellows joint. All the above-core structures are dismounted in the event of refuelling of the reactor.

Steam generators

The reference design of PIUS is based on utilisation of straight, once-through steam generators since these have a number of beneficial characteristics; the secondary side water inventory is small which reduces the total sub-cooling in the event of a large steam line break, normal operation with super-heated steam reduces the required span of primary side pressure variations during load follow operations, and they make it possible to reduce the containment volume somewhat. Utilisation of conventional U-tube steam generators has also been analysed, however, and their adoptation does not represent any major hurdles.

The steam generators are arranged vertically on two sides of the prestressed concrete vessel monolith, with the hot leg piping connected to the top.

Reactor coolant pumps

The reactor coolant pumps are sized-up versions of the wet motor, glandless pumps that are utilised as recirculation pumps in the ABB Atom BWR plant designs, in particular the design versions BWR 75 and BWR 90 with internal pumps. The pumps are mounted below the steam generators and structurally integrated with them; the motor housing represents an extension of the steam generator casing and is part of the reactor coolant pressure boundary, and the pump impeller is located in the steam generator outlet plenum.

The wet motors are three phase asynchronous motors supplied with power from individual "variable frequency - variable voltage" static frequency converters; this provides the desired speed control capabilities - continuous and controlled operation between 20 and 100% speed. The vertical bearings at the lower end of the motors are of hydrodynamic type and require a certain minimum speed to function properly; the 20% speed limit for continuous operation ensures an adequate bearing function.

Pressurizer

As noted above, the pressuriser proper with a steam volume and a pool of hot water (the pressuriser pool) is integrated into the upper steel vessel part of the reactor pressure vessel, and it does not constitute a separate component. The pressurizer is connected to the reactor pool via funnels up into the steam volume, and to the reactor primary loop via open passages from the pressuriser "pool".

Pressurisation of the reactor system is accomplished by means of steam supplied from an electrically heated recirculation boiler, drawing water from the water volume of the pressuriser (the pressuriser pool). The steam volume of the pressurizer is comparatively large and, together with its pool of saturated water,

the reactor system can accommodate pressure and level variations that may occur during operational transients and most accident situations. A pressuriser spray system using coolant from the discharge side of the coolant pumps, has also been provided, in order to avoid actuation of safety valve(s) in the pressure relief system in some special cases involving heat-up of the primary water mass and subsequent volume expansion. The primary loop water volume inventory control is also engaged in supporting this function, by ordering increased withdrawal rates of primary loop water.

Main coolant lines

The diameters of the hot and cold leg pipes are 665 mm and 635 mm, respectively. The pipes are made up of forged and machined sections of 304 stainless steel that are welded together.

The cold leg inlet nozzles are elongated to give room for a nozzle with reduced diameter section as a break flow limiter. The kinetic energy in the water is recovered in a diffuser before the flow enters the downcomer annulus inside the vessel.

6.3.2.5 Reactor auxiliary systems

Chemical and volume control system

Apart from the reactivity control tasks described below, the chemical and volume control system (CVCS) of PIUS supplies the primary loop with cleaned, filtered makeup water and with chemicals for water chemistry control. The primary loop water volume is controlled by withdrawing primary loop water and conveying it to the reactor water clean-up system for treatment. The inlet and withdrawal rates are controlled in such a way that the water level in the pressurizer is maintained at a "constant" position. The reactor water clean-up system also serves to clean and control the reactor pool water; a certain amount of pool water is continuously withdrawn, treated in the clean-up system and re-injected into the pool, possibly with chemical additives. The ratio between in- and outlet flow rates is adjusted when needed to maintain the hot/cold water interface in the upper density lock at a nearly constant position.

6.3.2.6 Operating characteristics

As noted above, reactor power is controlled by the boron content and temperature of the reactor coolant. During normal plant operation, the reactor power is controlled without adjustment of the boron content in the reactor coolant, by utilizing the strongly negative moderator temperature reactivity coefficient. A power change is accomplished by simply adjusting the feedwater flow rate (or the steam flow rate). An increase in flow rate results in a reduced temperature of the return flow to the reactor, a lowered average moderator water temperature and thus an increase in reactor power. This procedure is applied over a 40% power range with a 20%/min rate of change in plant power. Beyond this range adjustment of the boron content is needed in order to keep the reactor core coolant outlet temperature within acceptable limits. The boron content is controlled by injecting distilled water (for power increase) or high boron content water (for power decrease), and withdrawing a similar amount of water, corresponding to the procedures in normal PWR plants.

The moderator boron concentration is used for slow reactivity changes and for establishing the upper limit of a reactor power control range. It is also used for rapid shutdown by opening scram valves that let borated reactor pool water into the primary loop -at the coolant pump suction.

6.3.3 Description of turbine generator plant system

6.3.3.1 Turbine generator plant

The reference turbine plant design that has been developed for the PIUS plant design, is similar to that of present-day LWR plants. The 4.0 MPa, 270°C steam from the PIUS NSSS is at a lower pressure and temperature compared with the steam supplied from standard present-day LWR plants, and therefore PIUS requires a somewhat larger size turbine than other modern LWR plants. The nominal power output of the unit will be 635-665 MWe depending on the site conditions.

The generator is a two-pole type turbo generator, designed for continuous operation with hydrogen as the cooling medium for the rotor and water as the cooling medium for the stator windings. Its rotor is directly coupled to the turbine.

The turbine exhaust flows to a condenser which has three shells, located under the low pressure turbine exhaust hoods. The condenser also accepts the exhaust flow from the feed pump drive turbines and, on startup, hot standby and turbine trip, flow from the main steam and bypass system.

6.3.3.2 Condensate and feedwater systems

The feedwater system consists of the main feed pumps, two high pressure feedwater heaters, and associated piping. There are two 60% turbine driven main feed pumps, drawing water from the deaerator. Capability is provided to recirculate to the deaerator. There are also two electrically driven auxiliary feed pumps to be used during plant startup when main steam to drive the main pumps is unavailable.

Feedwater flow control is achieved by adjusting feed pump speed and the feedwater flow control valves. Feed pump speed is adjusted by modulating steam flow to the feed pump turbines. Extraction steam for the deaerator and high pressure heaters is provided from high pressure turbine extraction points, and the low pressure heaters are supplied from the low pressure turbines.

6.3.3.3 Auxiliary systems

Leakages and drains are collected in a liquid waste system that is designed to permit maximum reuse of water in a simple process; most of the collected water is chemically pure and may be reused as processed demineralized water after treatment in filters and ion exchangers. Excess water and "unclean" water is discharged, if its "quality" is acceptable, i.e., it has low radioactivity, and low content of other unacceptable products; otherwise, it is passed through an evaporator. The cleaned water can then be reused or discharged; evaporator residues are conveyed to the solid waste system.

The solid waste system comprises equipment for handling, sorting and compacting low level waste and for solidifying medium level waste originating from the plant, e.g., evaporator residues, ion exchanges and filter aids, always with an ambition of achieving small waste volumes. There is also an offgas system for treatment (delay and filtering) of potentially radioactive gases before releases to the atmosphere".

6.3.4 Instrumentation and control systems

6.3.4.1 Design concepts including control room

The instrumentation and control (I&C) system comprises the following main parts:

- Systems (or functions) needed for supervision and control of the normal operation of the plant;
- Systems (or functions) related to protection of plant components and systems;
- Systems for management of the core operation.

The I&C system is based on programmable technology and equipment. The different I&C functions are performed by various types of micro-processor or computer systems.

Data acquisition for process information, and the interface to process actuators, utilizes the simplest types of microprocessors, whereas the top level of the I&C system hierarchy uses powerful minicomputers, eg., for core calculations. Intermediate types of microcomputers are utilized for control and operation, logic and signal treatment.

A separate command centre at the site will communicate with the computers of the I&C system to permit personnel in this centre to follow operations in the plant, but not affect it, with one exception: initiation of a reactor scram.

The I&C system is structured in a systematic way with different levels and sections in the hierarchy. Some considerations of the hierarchy include:

- relationships within the different plant system groups, e.g., reactor, turbine, electric power distribution;
- responsibilities of the operator(s) in the control room;
- need for redundancy, separation and reliability;
- degree of automation;
- testability and test requirements.

The main control room is situated in the control building, in the vicinity of the reactor building. The design basis of the control room is that all supervision and control of the systems needed during normal operation can be executed from this centre. The I&C system is furthermore designed to require only one operator in the control room for normal operation. Current safety rules, however, generally require two operators.

Depending on the shift organization at a specific plant, there may be one or more operator desks. The basic PIUS arrangement includes two desks, one for the reactor operator and one for the BOP operator. Each desk is equipped with a number of colour VDUs with keyboards and tracker balls.

The displays on the VDUs are structured systematically to present:

- Overviews and trends
- Alarms
- Means for manual control.

Events or long alarm lists can be presented on the VDUs, but they are normally printed out in the adjacent recording room. Manual orders to process actuators and electric breakers are initiated by a sequence of keyboard and tracker ball actions. Normally, two manual actions are required for initiating a process actuation.

The shift supervisor has a dedicated desk, with VDUs for presentation of plant information (generally of overview character). Manual control of the plant processes cannot be performed from this desk. The shift supervisor desk includes equipment for communications inside the plant, as well as equipment for offsite communications.

The control room is also provided with an overall display panel which is equipped with conventional display units; the information displayed on this panel can be observed within the whole main control room area.

6.3.4.2 Reactor protection system and other safety systems

The PIUS plant is provided with instrumentation systems, protection, logic, and actuation systems for reactor shutdown, residual heat removal, containment isolation, etc. in a similar way as present-day LWR plants. Their importance for ensuring safety is significantly reduced in a PIUS plant. The equipment of these instrumentation, monitoring, protection, and actuation systems is separated from that of other systems and located in separate, physically well protected compartments at the bottom of the reactor building. The reactor protection system (RPS), with a two-out-of-four coincidence logic, has the task of initiating power level reduction, reactor shutdown or reactor scram when reactor process parameters exceed set limits, in order to prevent further departure from permissible conditions.

Compared with current commercial LWR designs a number of safety-grade systems have been eliminated, allowing major simplification of the plant; the control rods and the safety injection boron system are replaced by the density locks, the automatic depressurization system is not required, the auxiliary feedwater supply system for RHR is replaced by the reactor pool, the containment heat removal and containment spray systems are replaced by the passive cooling of the reactor pool. The safety-grade closed

cooling water system, HVAC systems, and a.c. power supply systems have been replaced by non-safety-grade systems.

The remaining safety-grade functions are performed by the reactor protection system which initiates opening of the scram valves to achieve a reactor scram, the containment isolation system which initiates isolation of the containment by closing isolation valves, the reactor vessel safety valves which is based on pressure-activated components, and the passive reactor pool cooling function. These functions are not needed for the protection of the core, however.

6.3.5 Electrical systems

6.3.5.1 Operational power supply systems

The plant auxiliary electric power supply systems have three distinguishable AC voltage levels; the medium voltage level (eg. 6 kV), at which the largest electric loads are connected; the low voltage level (eg. 660 V), at which the major number of plant process loads are connected; and the battery-backed low voltage (eg. 400/240 V), at which the computer systems and various control equipment are connected. Basically, there is no general DC distribution in the plant; the control equipment needs of DC are met by local rectifier units supplied from the battery-backed AC systems.

Compared with present-day LWR plants, the electric power supply systems of PIUS have been simplified significantly. The main reason for that is obviously the "inherent" self-protective functions of PIUS that eliminate many traditional safety-grade systems or reduce their importance to such extent that they can be declassified. Thus, a "two train" electrical supply system structure has been found fully acceptable with respect to safety and plant availability concerns. In order to improve the reliability of the power supply for important process functions, e.g. for protection of the capital investment, low voltage diesel generators are provided in the PIUS plant, but they are not safety-grade. The elimination of most of the DC distributions represents another major simplification of the electrical power supply system.

6.3.5.2 Safety-related systems

There are no safety-grade diesel generating units in the design. Power supplies to the "remaining safety-grade" equipment are provided by separated batteries and DC distributions in the reactor building.

6.3.6 Safety concept

6.3.6.1 Safety requirements and design philosophy

The main emphasis in the development work has been to prevent core degradation accidents under any credible conditions, without recourse to the function of safety equipment needing actuation signal or power or to operator actions or interventions; in other words, in an entirely passive way. The economy and operability of the plant must not be sacrifized to achieve this, however.

The primary goal in nuclear safety is to prevent radioactive matter from entering the environment and unprotected parts of the plant premises. By far the dominating part of such matter, and practically all the volatile nuclides that are of real concern in this context, are located in the reactor fuel inside the fuel element cladding.

Hence, protection of the core against damage is the top-level goal in reactor safety. This means that the temperature of the cladding must be kept sufficiently low at all times, which, in practice, can be ensured by fulfilling the following two conditions:

1. Keep the core submerged in water at all times.
2. Keep the rate of core heat generation below the cooling capability of the surrounding water [avoid Departure from Nucleate Boiling (DNB)].

The Defence-in-Depth approach has been, and will remain, an important principle in the nuclear safety strategies. It is, of course, applied also for PIUS - but with a significant shift in emphasis towards prevention/protection, and a corresponding relaxation with respect to requirements on accident management, in particular taking into consideration that (so far) no accident sequences leading to core damages have been identified.

The Design Basis Event (DBE) approach to the deterministic analysis of PIUS plant safety is essentially the same as that for present day LWR plants. The most immediate model has been that of the US standard ANSI/ANS-51.1, to some extent supplemented by ANSI/ANS-52.1. The acceptance criteria have had to be modified somewhat, however, since the design of PIUS does not quite lend itself to the events and terminology of the ANSI standard.

The scope of events analyzed goes far beyond the traditional set of DBEs, however, and includes also events and sequences of events that are normally ignored.

The PIUS design incorporates inherent self-protection against severe core damage, and utilisation of a mechanistic source term for the analysis of environmental impacts appears appropriate. The releases of radioactive matter to the environment will be very small after any accident, and there is no need for off-site emergency plans of the same kind as for present day LWR plants.

Accident analyses performed so far confirm that the safety goals are fulfilled. No accident sequence leading to core degradation has been identified. The safety performance can be illustrated by the response to a hypothetical large LOCA, a double-ended cold leg pipe rupture at a low location.

The hot leg pipe outflow stops when the water level in the vessel has dropped below the hot leg nozzle, and pressure equilibrium between the containment and the reactor vessel is established. The siphon breaker arrangement provides "containment" pressure also on the inside of the cold leg nozzle, and the large outflow from the reactor system stops - all by itself. The core is cooled by reactor pool water in natural circulation, and the decay heat is absorbed in the pool. The pressure in the containment attains a peak of about 270 kPa after about 1 minute, and then decreases due to steam condensation on containment walls and structures. In about 2 hours, it is down to slightly above atmospheric pressure again, due to steam condensation on containment walls and structures.

The reactor pool is cooled by the passive system arranged in four groups, each with a cooling tower on top of the reactor building. Postulating failure of one group, the reactor pool water temperature will still be kept below boiling temperature at atmospheric pressure.

This accident does not result in fuel damages, and the release of radioactive material to the containment is determined by the amount of such products present in the water prior to the accident. The iodine "spiking" will remain in the reactor water since there will be no boiling. Hence, release to the containment will be very small, and taken its moderate and short pressurization the release to the environment will be minimal; the whole body dose at the plant fence has been calculated to about 1 mrem. It may be noted that a significant reduction of requirements with respect to emergency preparedness has been a design objective for PIUS.

In addition to the deterministic analyses, and the simulations performed with thermal-hydraulic computer codes, a preliminary Level 1 PSA study has been completed in a joint effort by ABB Atom and the Italian power company ENEL with support from ANSA (formerly ENEA-DISP), the regulatory body of Italy. The study represents a first comprehensive review of the PIUS plant design, based on ultra-conservative assumptions - the failure frequency for the prestressed concrete vessel ended up as somewhat higher than for a steel vessel, and a number of transients were just postulated to yield core damage even though calculations showed that they would not. Still, the resulting "core damage" frequency is below 10^{-7}.

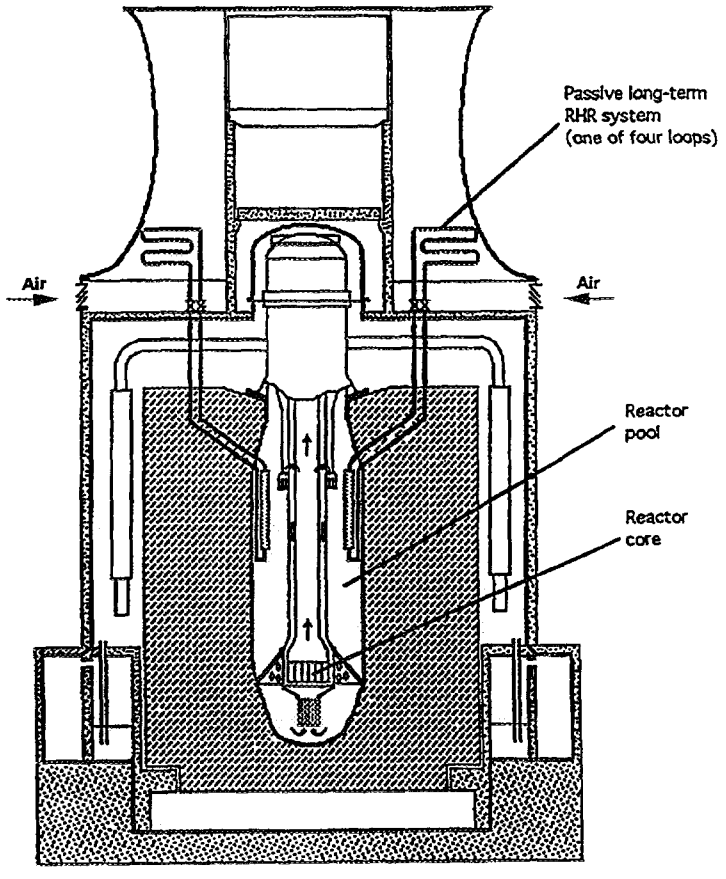


Figure 6.3-4 PIUS - Safety-grade structures

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

Essentially, PIUS is a PWR that predominantly utilizes existing LWR technology. The major design parameters have been conservatively chosen:

- Lowered core power density;
- Lowered linear heat rating of the fuel;
- Negative power reactivity coefficient throughout the operating cycle;
- Lowered reactor pressure and temperature as compared to present-day PWR plants

Essential departures from current LWR technology are limited to the following areas:

- Thermal-hydraulic principle of the reactor
- Density locks (thermal barriers), siphon breakers, and wet thermal insulation
- Prestressed concrete reactor vessel
- Long-term passive residual heat removal system
- Reactivity control without control rods

The self-protective thermal-hydraulics have been successfully demonstrated in normal and under severe transient conditions. The remaining departures from current reactor technology listed above, except the absence of control rods, have been either verified through testing or have a sound basis in technology outside of reactor technology. The absence of control rods is actually an advantage since mechanical devices and interacting detector and insertion systems are eliminated. The risk of serious reactivity insertion due to control rod malfunctions is also eliminated.

The protection against core degradation accidents is ensured by the laws of physics alone, and all essential safety functions for this are located in or on the safety-grade structures (Figure 6.3-4): the concrete vessel with its cavity; the containment; and the cooling towers on top of the containment.

As noted above, the PIUS design is based on established LWR technology, and the existing regulatory framework should be sufficient as a basis for licensing. From a licensing point of view, the concrete vessel and the absence of control rods represent important departures from current technology, but they are also, together with the totally passive safety systems, the key elements for the favourable safety performance.

Safety systems configuration

The reactor core is physically well protected by the enclosing containment structure and the thick-walled, strong concrete vessel walls, and an "ultimate" protection against overheating and fuel damage is provided by the unique PIUS arrangement - with the core submergence in a large pool of borated water, the ever-present openings between the pool and the primary loop, and transition to reactor shutdown and core cooling in a natural circulation mode without reliance on equipment for detection of off-normal conditions, initiation of actions, actuation of equipment, nor equipment relying on the displacement of mechanical bodies. Intervention of active systems is needed to keep the reactor in operation, not for safety, preventing it from reverting to a state of shutdown and natural circulation core cooling.

PIUS is provided also with instrumentation, protection logic, and actuation systems for reactor shutdown, residual heat removal, containment isolation, etc. in a similar way as present-day LWR plants, but their importance for safety is significantly reduced, however. The equipment of these instrumentation, monitoring, protection, and actuation systems is separated from that of other systems and located in separated compartments at the bottom of the reactor building. The reactor protection system (RPS), with a two-out-of-four coincidence logic, has the task of initiating power level reduction, reactor shutdown or reactor scram when reactor process parameters exceed set limits, in order to prevent further departure from permissible conditions.

In most cases, a runback to a lower power level, using the secondary side control, or going to hot standby or hot shutdown conditions by injecting high boron content water into the primary loop, will be an adequate countermeasure. A reactor scram is initated only in a few accident situations by opening the scram valves which will let borated reactor pool water into each of the cold legs (at the suction of the coolant pumps). Borated water then reaches the core in a few seconds and shuts down the reactor to hot, subcritical conditions; primary loop structures will be subjected to a rapid cool-down by some 50-60K - a rather mild thermal transient and quite insignificant with respect to thermal fatigue.

The scram valves system is considered safety-grade, even though the system function does not fully comply with the requirements on safety-grade systems; its successful function depends on continued operation (at least for a certain period of time) of the non-safety-grade main coolant pumps. However, whenever these pumps stop operating, the reactor will immediately be shut down by the self-protecting shutdown mechanism, - by borated pool water ingress through the lower density lock.

The inflow of cold pool water results in a displacement of warm primary loop water to the pool, i.e., heat is transported from the primary loop to the pool. The water in the pool is cooled by two systems, one active, non-safety-grade system and one passive, closed loop safety-grade system. The latter comprises (Figure 6.3-5) a number of water-water coolers along the perimeter of the concrete vessel cavity at the level of the upper density lock.

These coolers are connected to air coolers in four natural draft cooling towers on top of the containment/reactor building; there is one 90° section at each corner, separated from each other by the reactor building superstructure in such a way that external impacts, e.g., from a crashing aircraft, should not damage more than one. The cooling circuits between the pool coolers and the cooling towers operate in an entirely passive way with natural circulation of the coolant. The water mass of the reactor pool is very large compared with the primary loop volume (about 4 times larger) and the pool water heatup will be limited; there is no need for an automatic depressurization system. The pool water ingress through the lower density lock basically replaces the control rods and the safety injection system of current commercial PWR designs, and the passive cooling of the reactor pool eliminates the need for containment heat removal and spray systems.

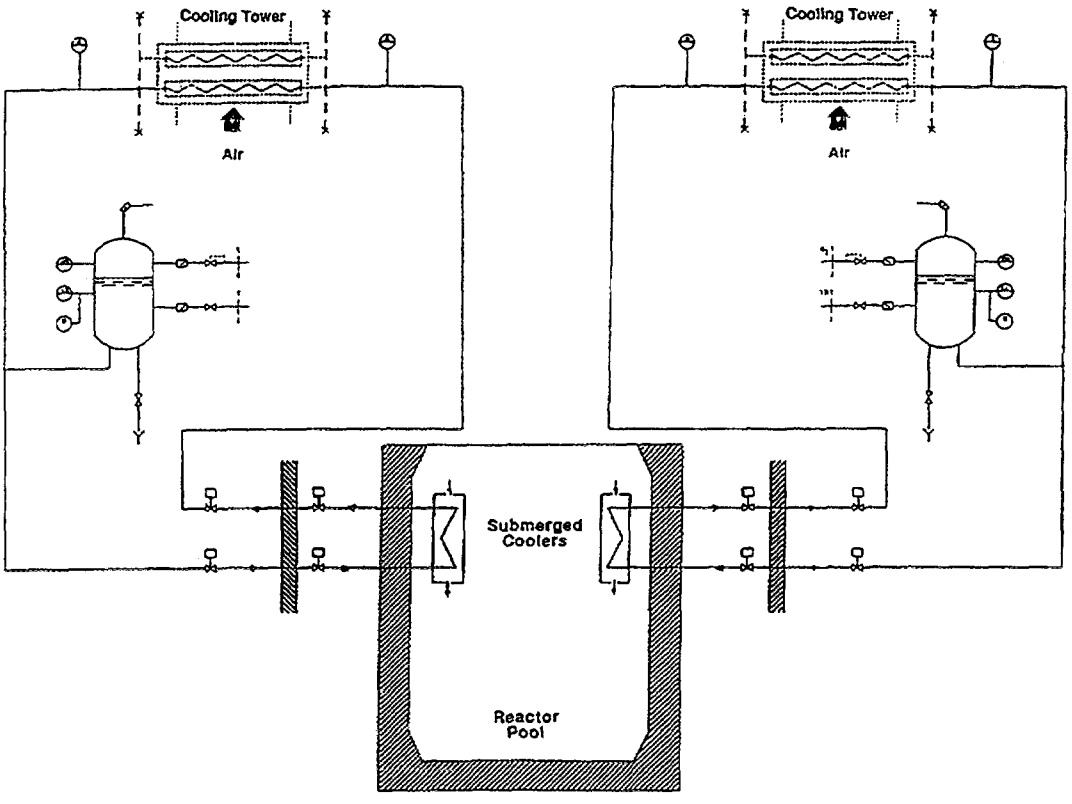


Figure 6.3-5 PIUS - Passive closed cooling system for heat removal from the pool

6.3.6.3 Severe accidents (beyond design basis accidents)

As noted (Section 6.3.6.1) the main emphasis in the development work has been to prevent core degradation accidents under any credible conditions, without recourse to the function of safety equipment needing actuation signal or power or to operator actions or interventions; in other words, in an entirely passive way.

6.3.6.3 Severe accidents (beyond design basis accidents)

As noted in Section 6.3.6.1, the main emphasis in the development work has been to prevent core degradation accidents under any credible conditions, without recourse to the function of safety equipment needing actuation signal or power or to operator actions or interventions; in other words, in an entirely passive way.

The scope of accident initiating events analyzed goes far beyond the traditional set of DBEs, and includes also events and sequences of events that are normally ignored, but the analyses performed by ABB Atom (and by Los Alamos National Lab. in the US) have not revealed any "realistically", conceivable accident sequences leading to core damages. Hence, the occurrence of a severe core damage accident is in practice ruled out by the design configuration.

6.3.7 Plant layout

6.3.7.1 Buildings and structures, including plot plan

The plant layout (Figure 6.3-6) features four main separated blocks of buildings to improve accessibility and facilitate parallel construction and assembly activities which help shortening the construction schedule. The plant has one entrance only for daily use, backed up by a second emergency exit.

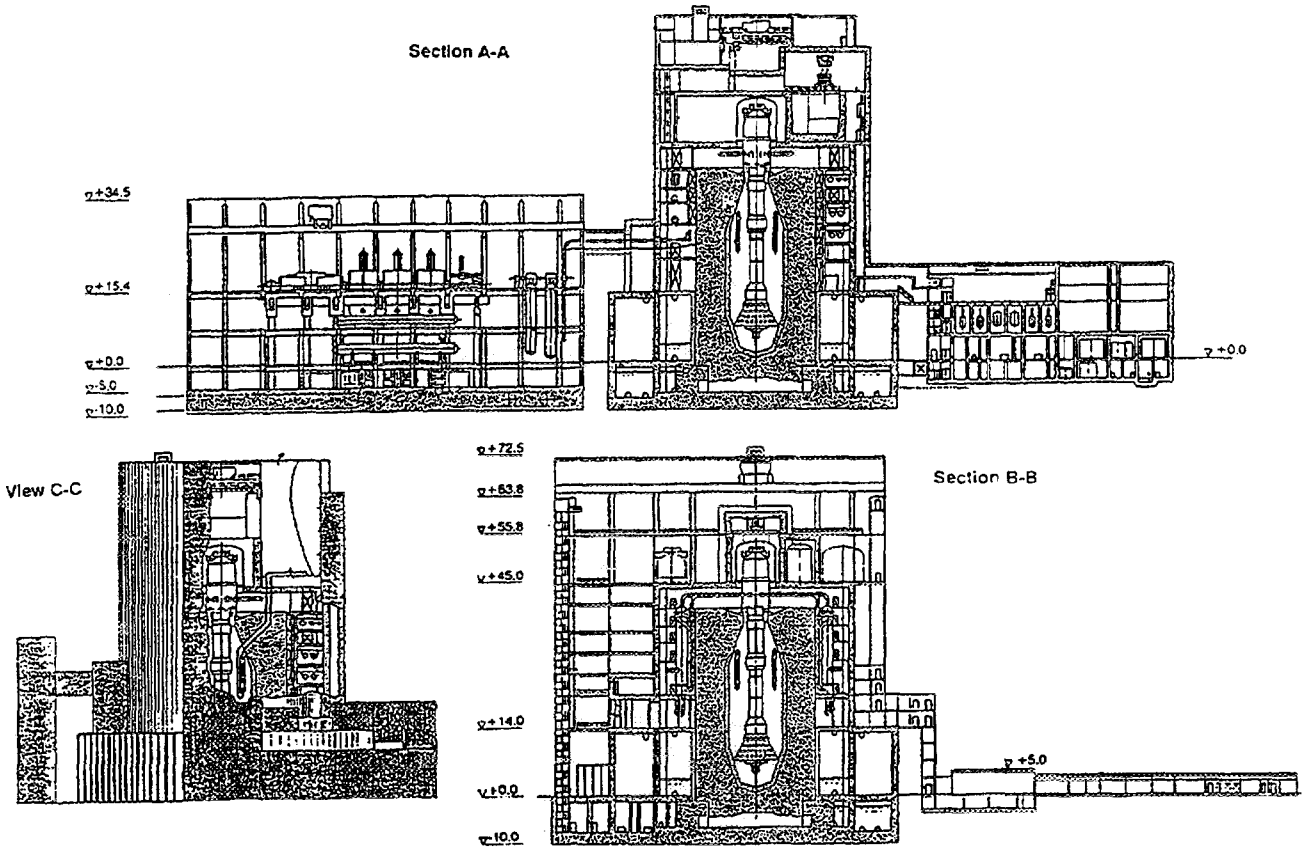


Figure 6.3-6 PIUS - General building arrangement

The main block is made up by the reactor building, basically a cylindrical structure with a diameter of about 60 m (decreasing to about 45 m about 15 m above grade), and a height above grade of 72 m. Vertical shafts are arranged on two diametrically opposite sides of the "cylinder; from a low level up to one of the reactor service room aisles. One of these shafts is the transport shaft (from the ground level), eg., for fuel transport to/from the plant, and the other (from the 14 m level) provides communication with the adjacent control building. The second block includes the reactor auxiliary and waste management building, housing the reactor water cleanup system and the liquid and solid radwaste systems, the radioactive maintenance shops, housing the active workshop, and storage rooms for potentially radioactive waste.

The control building, with the main control room, computer rooms, personnel entrance, etc., and the diesel generator and non-vital low voltage switchgear building make up the third block.

The fourth block, finally, is formed by the turbine building, the non-vital medium voltage switchgear building, the transformer enclosures, the service water pump house and the circulating water pump house. This block is located on the other side of the reactor building, compared with the second block.

The layout is divided into clean and potentially contaminated areas with directional ventilation, where air from potentially contaminated areas could leak to cleaner areas. Filtered ventilation by way of the stack is available for potentially contaminated rooms when needed. Electrical systems and process systems are separated from each other and located in different rooms and culverts. Process systems are similarly split into radioactive or non-radioactive systems.

Only the systems that are part of the high pressure reactor coolant system are located within the containment. Systems carrying hot pressurized reactor water are not allowed to extend beyond the containment. The reactor water cleanup and the liquid and solid waste handling systems are located in a separate building with concrete walls for separation and shielding of major components.

The building arrangement is also characterized by a system of communication routes for personnel and equipment, between and inside buildings, that serves to facilitate maintenance, inspection and repair work by ensuring good accessibility to plant equipment together with a suitable design and installation of the process systems, a meticulous choice of materials, a proper routing of ventilation air flows, this paves the ground for achieving low operational radiation exposure, as demonstrated by the ABB Atom BWR 75 plants in operation in Scandinavia; they have consistently been operated at an annual occupational exposure of 1 mansievert or lower.

PIUS builds on the experience and know-how from design, construction and operation of the BWR 75 plants. The same installation and ventilation principles are maintained, the accessibility considerations are applied, and very stringent material specifications are adopted; hence, low occupational exposures are anticipated also for the PIUS plants.

As noted above, the upper portions of the reactor building, which constitute the physical protection for the reactor cavity and the spent fuel storage pool, and the reactor containment is designed to withstand the impact of a crashing airplane. And the reactor building complex, including the enclosed reactor containment and the safety-grade equipment, is designed against the effects of earthquakes. The reference design safe shutdown earthquake (SSE) has been set to 0.3 g.

6.3.7.2 Reactor building

The reactor building is the largest building of the plant. Basically, it is a cylindrical structure with a diameter of about 60 m (decreasing to about 45 m about 15 m above grade), and a height above grade of 72 m. Vertical shafts are arranged on two diametrically opposite sides of the "cylinder"; from a low level up to one of the reactor service room aisles. One of these shafts is the transport shaft (from the ground level), e.g., for fuel transport to and from the plant, and the other (from the 14 m level) provides communication with the adjacent control building.

The reactor service room at the top of the building has a second aisle, oriented perpendicularly to the first one. The natural-draft cooling towers for the long-term passive RHR system are located in the quadrants between these reactor service room aisles, i.e., the four cooling tower are physically protected by the reactor service room structures.

All safety-grade systems in the PIUS plant are located within the reactor building which encloses the containment, the fuel handling equipment, the fresh fuel storage, the spent fuel storage pool and the emergency control room (the auxiliary shutdown facility) with associated instrumentation, control equipment and batteries for electric power supply.

6.3.7.3 Containment

The nuclear steam supply system (i.e., the concrete vessel and the reactor system) is, in a similar way as other LWRs, enclosed in a large containment structure. The reference design containment is of pressure suppression type. Blowdown pipes lead from the drywell into a large condensation pool in the wetwell. All equipment containing reactor loop or reactor pool water at high pressure and high temperature is located inside the containment, which is designed to withstand a double-ended break of the largest pipe. The structure is made of reinforced concrete with a strength capable of resisting the impacting of a crashing aircraft. The whole containment is provided with a steel liner in order to ensure leaktightness. A steel dome closes the shaft above the reactor vessel.

During refuelling operations the containment dome and the reactor vessel head are removed, and the cavity above the dome is filled with water. The reactor internals are lifted out in sections, and placed in the water-filled cavity. The refuelling is carried out with a conventional refuelling machine from the reactor service room. Fresh fuel is brought into the cavity from a fresh fuel storage in the reactor building, and the spent fuel is removed to an adjacent spent fuel pool at the reactor service room floor level.

The steam lines from the steam generators and the feedwater lines to them are provided with isolation valves inside and outside the containment wall - the outer valves being located in a separate protected compartment. The pressure relief valves on the steam lines blow to the condensation pool inside the containment, as do the pressure relief valves of the reactor pressure vessel.

The containment structure is designed to withstand a double-ended break of the largest pipe - the dominant internal event with respect to pressurization of the containment vessel. The design case is a break of the cold leg close to the main coolant pump outlet, combined with a "failure to scram" and loss of all AC power (station blackout) for hours. Considerable margins are therefore available in the design. In most severe accidents (beyond design basis accidents) situations, the stresses and loads on the containment structure will not exceed those occurring in the design case.

6.3.7.4 Turbine building

The turbine building which houses the main plant turbine and the majority of the turbine systems, is a conventional building of about 35 m height above grade with a length and width of about 80 m and 50 m, respectively. It is located on one side of the reactor building, at a distance of some 5 m, with the turbine shaft projecting in the direction from the reactor building. The steam and feedwater lines are run through a communication culvert between the two buildings at an elevation of about 25 m. The turbine building is not part of the plant controlled access area and the leaktightness requirements are not set by nuclear criteria.

6.3.7.5 Other buildings

The second block of the central building compound is located on the side of the reactor building that is opposite to the turbine building. It includes the reactor auxiliary and waste management building, which houses the reactor water cleanup system and the liquid and solid radwaste systems, the radioactive maintenance shops, which houses the active workshop, and storage rooms for potentially radioactive waste.

The control building, with the main control room, computer rooms, personnel entrance, etc., and the diesel generator and non-vital low voltage switchgear building make up the third block, which is located adjacent to the reactor building, on one of the free sides. It is connected with the reactor building by means of a communication shaft up to the reactor service room.

The non-vital medium voltage switchgear building, the transformer enclosures, the service water pump house and the circulating water pump house are located at the sides of the above-mentioned turbine building, making up the fourth block.

Water treatment building, raw water treatment building, storage and workshop buildings, security building, meteorology tower, main plant stack and high voltage switchyard structures represent examples on further buildings and structures that will be found on the plant site.

6.3.8 Technical data

General plant data		
Power plant output, gross	650	MWe
Power plant output, net	640	MWe
Reactor thermal output	2 000	MWt
Power plant efficiency, net	32	%
Cooling water temperature	15	°C
Nuclear steam supply system		
Number of coolant loops	4	
Primary circuit volume, including pressuriser	650	m^3
Steam flow rate at nominal conditions	1 012	kg/s
Feedwater flow rate at nominal conditions	1 012	kg/s
Steam temperature/pressure	270/4.0	°C/MPa
Feedwater temperature/pressure	210/4.5	°C/MPa
Reactor coolant system		
Primary coolant flow rate	13 200	kg/s
Reactor operating pressure	9.5	MPa
Coolant inlet temperature, at RPV inlet	260	°C
Coolant outlet temperature, at RPV outlet	289.3	°C
Mean temperature rise across core	29.3	°C
Reactor core		
Active core height	2.50	m
Equivalent core diameter	3.75	m
Heat transfer surface in the core	5 000	m^2
Fuel inventory	80.6	t U
Average linear heat rate	11.9	kW/m
Average fuel power density	24.8	kW/kg U
Average core power density (volumetric)	72.5	kW/l
Thermal heat flux, F_q	400	kW/m^2
Enthalpy rise, F_H		

Fuel material	Sintered UO_2	
Fuel assembly total length	2 820	mm
Rod array	square, 18×18	
Number of fuel assemblies	213	
Number of fuel rods/assembly	312	
Number of control rod/instrumentation guide tubes	0 / 4	
Number of spacers	-	
Enrichment (range) of first core [average]	2.0	Wt%
Enrichment of reload fuel at equilibrium core	3.5	Wt%
Operating cycle length (fuel cycle length)	12	months
Average discharge burnup of fuel	45 500	MWd/t
Cladding tube material	Zr 4	
Cladding tube wall thickness	0.6	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly	542	kg
Active length of fuel rods	2 500	mm
Burnable absorber, strategy/material	adv. appl. of Gd_2O_3 in fuel	
Number of control rods	0	
Absorber rods per control assembly	0	
Absorber material	NA	
Drive mechanism	NA	
Positioning rate	NA	
Soluble neutron absorber	Boron	
Reactor pressure vessel		
A. Prestressed Concrete Reactor Vessel (PCRV)		
Overall height (without the vessel upper part)	44	m
Overall width	26.8×26.8	m
Inside diameter (max cavity diameter)	12.2	m
Cavity volume (to top of concrete)	3 300	m^3
Net reactor pool water volume (above top of core)	2 300	m^3
Wall thickness (minimum)	7.4	m
Stainless steel liner thickness	15	mm
Total weight (without water)	63 000	t
Design pressure/temperature	10.5/315	MPa/°C

B. Pressure Vessel Upper Part (PVUP)

Overall height with the head	14.4	m
Overall width	9	m
Inside diameter	6.1	m
Wall thickness	180	mm
Base material:	pressure vessel steel, SA 533 B	
Stainless steel cladding thickness	10	mm
Inside diameter of hot/cold leg nozzles	665/635	mm
Design pressure/temperature	10.5/315	MPa/°C
Transport weight (including RPV head	550	t

Steam generators

Type	Vertical, straight, once-through	
Number	4	
Heat transfer surface	24 000	m^2
Number of heat exchanger tubes	8 000	
Tube dimensions	14.2/15.9	mm
Maximum outer diameter	2 500	mm
Total height	19 000	mm
Transport weight	103	t
Shell and tube sheet material	SA 533, Class 2, Grade B/SA 508, Class 3a	
Tube material	Inconel 600	

Reactor coolant pump

Type	Variable speed, wet asynchronous motor & glandless shaft centrifugal pump	
Number	4	
Design pressure/temperature	10.5/315	MPa/°C
Design flow rate (at operating conditions)	3 300	kg/s
Pump head	80	m WG
Power demand at coupling, cold/hot	2 500	kW
Pump casing material	SA 533,Class 2, Grade B	
Pump speed [at 100% reactor power]	1 450	rpm

Pressuriser

Total volume	150	m^3
Steam volume: full power/zero power	100/	m^3
Design pressure/temperature	10.5/315	MPa/°C
Heating power of the heater rods	350	kW
Number of heater rods	8	
Inner diameter [PRZR is integrated in RPV]	6 100	mm
Total height	NA	mm
Material	cf RPV upper part	
Transport weight	NA	t

Pressuriser relief tank — Not applicable

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Pressure-suppression	
Overall form (spherical/cyl.)	cylindrical	
Dimensions (diameter/height)	63/61	m
Free volume (drywell/wetwell/cond.pool)	20 000/20 000/2 000	m^3
Design pressure/temperature (DBEs)	0.5/	kPa/°C
(severe accident situations)	0.5/	kPa/°C
Design leakage rate	< 1	vol%/day
Material	Reinforced concrete with steel liner	
Is secondary containment provided?	yes	

Reactor auxiliary systems

Reactor water cleanup,	capacity, primary/pool loop	2×6/2×6	kg/s
	filter type	deep bed, radial flow type	
Residual heat removal,	at high pressure [via CVCS]	2×6	kg/s
	at low pressure [via the pool]	2×250	kg/s
Coolant injection,	at high pressure [via CVCS]	2×6	kg/s
	at low pressure [via the pool]	as needed	kg/s

Power supply systems

Main transformer,	rated voltage	400/20	kV

	rated capacity	800	MVA
Plant transformers,	rated voltage	20/6.9	kV
	rated capacity	35/25/25	MVA
Start-up transformer	rated voltage	110 (or 70)/6.9	kV
	rated capacity	25/15/15	MVA
Medium voltage busbars (6 kV or 10 kV)		2×6.6	kV
Number of low voltage busbar systems		2×660, 2×400	V
Standby diesel generating units: number		2	
	rated power	1.2	MW
Number of diesel-backed busbar systems		2	
Voltage level of these		660	V ac
Number of DC distributions		2	
Voltage level of these		110	V dc
Number of battery-backed busbar systems		2	
Voltage level of these		400	V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	Double-flow, condensing	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/ 3 LP	
Turbine speed	3 000	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	3.87/270	MPa/°C

Generator

Type	3-phase, synchronous, turbogenerator	
Rated power	785	MVA
Active power	667	MW
Voltage	20	kV
Frequency	50	Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type	3 shells, single or double flow	
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate	31.8	m^3/s
Cooling water temperature	15	°C
Condenser pressure	50	hPa

Condensate pumps

Number	3×50%	
Flow rate [per pump]	320	kg/s
Pump head	1.43	MPa
Temperature		°C
Pump speed		rpm

Condensate clean-up system

Full flow/part flow	Full condensate flow
Filter type	precoat, deep-action

Feedwater tank

Volume	300	m^3
Pressure/temperature		MPa/ °C

Feedwater pumps

Number	2×60%	
Flow rate [per pump]	~600	kg/s
Pump head	4.6	MPa
Feedwater temperature	210	°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages	3 LP + FWT + 2 HP
Redundancies	2×50% LP and HP heaters

6.3.9 Project status and planned schedule

ABB Atom has been working on PIUS type reactors for more than a decade with considerable detailed design and analyses performed for a number of design versions. Based on these activities and inputs from utilities and others a promising and mature design concept has evolved in which demonstrated component technology is utilized to the maximum extent. The novel features have been sufficiently studied to eliminate concerns regarding the technical feasibility and practicability of the concept, especially from the point-of-view of safety and operability. Additional testing is considered necessay, however, to provide information and data to support the detailed design and arrangement for a commercial plant; further hydraulic tests to support computer codes and large scale "density lock" tests will be carried out as soon as possible, followed by "component-related" tests and a large scale integrated system test.

PIUS was for some time discussed with the Italian state utility ENEL (since July 1992 a private company) for an evaluation and assessment of the practical feasibility, and the Consorzio PIUS, formed by ABB Atom together with the Italian companies ANSALDO and FIAT, in June 1992 presented an offer to ENEL on a joint design study for adaptation to Italian conditions and requirements; no contract agreement was reached, however, due to the privatization. A feasibility study has been performed in the People's Republic of China. Efforts have been made in the USA, together with ABB Combustion Engineering Nuclear Power, for marketing the design and for having it reviewed and licensed by the US NRC; an NRC pre-licensing review has been going on, with an SER originally scheduled for the Spring of 1994, but recent budget cuts have resulted in an abortion of the review.

Making a classification of the project status, with respect to design completion, represents a very difficult task. It may be stated that the Conceptual Design has been established, and that portions of the typical activities of the Basic Design are well underway; i.e., the status of the PIUS project lies somewhere between the Conceptual and Basic Design states.

As noted above, the basic R&D work, to prove the feasibility and practicality, has been completed. Some supplementary testing, to support the detailed design, is planned, but for the time being, put on hold, awaiting additional external support for continuation of the development activities. Certain "component"-related tests and a large-scale integrated system test, for final verification before start of construction, are envisaged for the future.

With respect to licensing activities, reference is made to the pre-licensing review by the US NRC; no formal licensing application has been submitted. Preliminary assessments of PIUS-type reactors have been made by STUK, the Finnish licensing authority, and by the GRS (Gesellschaft für Anlagen und Reaktor-sicherheit), a German Advisory Institute to the Government.

Construction of a PIUS plant involves a few major items that are important for the critical path; the prestressed concrete reactor vessel (PCRV) together with the containment part of the reactor building, and the reactor pressure vessel upper part (the steel extension). The construction activities have been analyzed by the team of civil engineering, installation and commissioning supervisory personnel that built and commissioned the Oskarshamn 3 Nuclear Power Plant in Sweden in 57 months from the first pouring of concrete to start commercial operation. The PIUS plant containment is similar to that in Oskarshamn 3, and the PCRV can utilize the same construction technique. This means that the planning team could draw on their own experience when establishing the schedule for the PIUS plant construction. The resulting schedule indicates a total construction time of 42 months for the plant -from pouring of the first concrete to start commercial operation, or 36 months to fuel loading. This time is probably conservative since the possibilities of reducing it by onsite or offsite prefabrication (or modularization) have not been taken into account.

Detailed turnkey cost estimates have been made for an nth of a kind 600 MWe PIUS plant (2000 MWt) and for a conventional ABB Atom Advanced BWR plant for 700 MWe output (2150 MWt), manufactured and constructed under Scandinavian conditions. These cost estimates indicate a 10% advantage for the PIUS 600 over the BWR 700 MWe in overnight cost per net kWe output. The lower steam pressure and

temperature of PIUS imply a lower thermal efficiency and thus a somewhat higher fuel cycle cost than for the BWR. On the other hand, the construction time is shorter and the personnel costs are anticipated to be lower thanks to the simplicity of the PIUS plant. As a result the total costs per energy unit (kWh) are also estimated to be lower than for the 700 MWe BWR plant.

A comparison has also been made with an 1100 MWe BWR plant and then the PIUS plant will be at a disadvantage of close to 10%; the economy of scale cannot be beaten completely - larger plant sizes will be more economical with respect to the kWh cost.

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6.4 SPWR, JAERI/IHI, JAPAN

6.4.1 Introduction

The SPWR (System-integrated PWR) design concept has been developed by the Japan Atomic Energy Research Institute (JAERI). It employs a new arrangement which can provide highly passive safety, easy operation/maintenance and economic competitiveness using experience and technology already obtained in the course of the existing LWR development.

The SPWR is an integral PWR which is characterized by having the complete primary circuit, - the reactor core, the main coolant pump, and the pressurizer, - as well as the steam generator encompassed in the reactor pressure vessel. The reactor pressure vessel is covered with a water-tight shell and is installed in a water-filled containment vessel. The water of the containment vessel is cooled by a heat pipe type passive cooling system.

The SPWR employs passive systems, for basic safety functions such as reactor shutdown, short-term decay heat removal (or safety injection), and long-term decay heat removal. Accident mitigation can be achieved by means of passive safety systems such as the Passive reactor shutdown system, Pressure balanced injection system, Containment water cooling system, and an active Automatic depressurization system.

A remarkable feature of the SPWR is that it utilizes poison tanks filled with highly borated water (boric acid water) in place of control rods for reactor shutdown. The water-filled containment vessel and the passive engineered safety systems are further remarkable features.

The SPWR project was completed in 1993.

6.4.2 Description of the nuclear systems

6.4.2.1 Primary circuit and its main characteristics

The SPWR concept is applicable for a wide output range, up to 600 MWe. The following description refers to a 600 MWe (1 800 MWt) power plant.

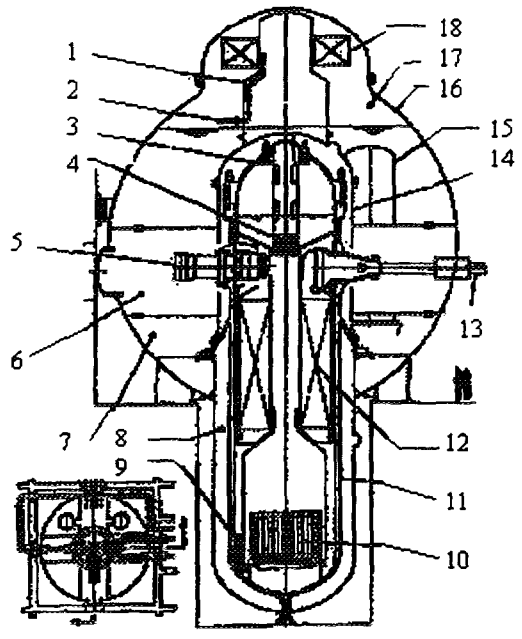
As described above, the SPWR adopts an integral type reactor system with all major primary coolant components installed inside the reactor pressure vessel (RPV); no large primary coolant piping is attached to the RPV. This way large break loss-of-coolant accidents (LOCA) are eliminated and safety enhanced.

The basic concept of the SPWR is shown in Figure 6.4-1. The reactor consists of the reactor pressure vessel (RPV), the reactor core, an integrated steam generator (SG), one canned motor main coolant pump (MCP), and an in-vessel pressurizer. The RPV is covered with a mirror-type thermal insulation enclosed by a water-tight shell, and installed in a water-filled containment vessel (CV). The water of the containment vessel is cooled by means of a heat pipe type passive cooling system.

6.4.2.2 Reactor core and fuel design

The reactor core (Figure 6.4-2) consists of 199 hexagonal fuel assemblies. Each fuel assembly consists of 325 fuel rods of which 30 are provided with burnable poison in a similar way as the well established practice in PWRs. The dimensions and composition of the fuel rods are the same as those of currently operating PWRs; i.e., the fuel pellets have a diameter of 8.19 mm, the cladding tubes are 9.5 mm in diameter with a wall thickness of 0.57 mm, the fuel rod pitch is 14.0 mm, and the fuel assembly pitch 259 mm.

The fuel assemblies are enclosed in perforated hexagonal channel boxes with a width of 258 mm. The average power density is relatively low, 65 kW/l.



1. PRESSURIZING CAP (FOR PBIS)
2. PBIS PRESSURIZING LINE
3. DEPRESSURIZING SPRAY HEADER
4. PRESSURIZER HEADER
5. MAIN COOLANT PLANT
6. DRYWELL (H2 GAS)
7. SUPPRESSION POOL (BORATED WATER)
8. DRYWELL(H2 GAS)
9. PRSS INJECTION LINE
10. CORE (HEIGH: 2.40 m, SQ. D: 3.83m) FUEL ASSEMBLIES
11. REACTOR VESSEL (INNER D: 5.5 m, WALL THICK: 285 mm, TOTAL HEIGH: 29 m)
12. STEAM GENERATOR
13. MAIN STEAM LINE
14. WATER-TIGHT SHELL
15. POISON TANK (X2) (BORATED WATER)
16. CONTAINMENT VESSEL (INNER D: 22m)
17. WETWALL (N2 GAS)
18. CONTAINMENT WATER COOLER (HEAT PIPE TYPE)

Figure 6.4-1 The SPWR concept (600 MWe)

The SPWR has no control rods and the reactivity is controlled by the core inherent characteristics and by chemical shim. The natural boron content in the primary coolant at the rated operating condition is 1 000 ppm at the beginning of the equilibrium cycle (BOEC) and 50 ppm at the end of it (EOEC). The enrichment of the reload fuel in an equilibrium core is 4.0 Wt% in the BA rods and 4.5 Wt% in the other rods for an operating cycle length of 24 months and an average discharge burnup of 48 GWd/t U.

6.4.2.3 Fuel handling and transfer systems

Since the upper part of the reactor riser is narrow (Figure 6.4-1) refuelling is performed by a special system which consists of a winch, an in-vessel manipulator and their control systems. A design study of this system was already made for the former SPWR concept and clarified its feasibility (Reference 2).

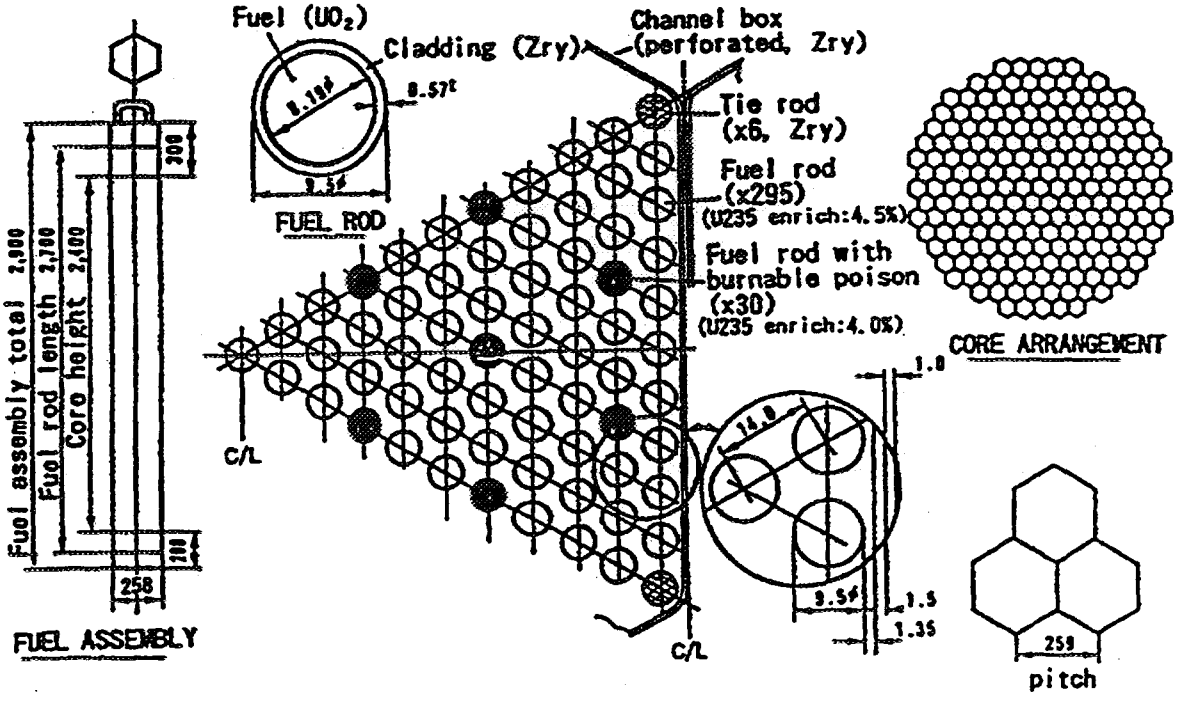


Figure 6.4-2 SPWR - Fuel and core concept

6.4.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel is relatively large in size since the SPWR is an integral type PWR. This provides a large primary water inventory and increases the distance between the reactor core and the RPV wall, thereby reducing the neutron fluence to the RPV. The technology for reactor pressure vessel fabrication is now well developed, and manufacturing of the RPV of the 600 MWe SPWR should not represent any major problem.

The RPV is enclosed in a water-tight shell which covers a thermal insulation of mirror-type, consisting of laminated thin stainless steel plates, to reduce heat losses from the RPV to the surrounding containment vessel water to a low level. During normal plant operation, the heat loss from the RPV to the containment vessel water is estimated to be less than 1 MW.

Reactor internals

No information provided.

Steam generators

The steam generator (SG) is of once-through helical coil type, located in the annular space above the core. It has 5 384 heat transfer tubes made of INCOLOY-800. The tubes are 19 mm in outer diameter and 2.0 mm in thickness.

Pressurizer

The pressurizer, which is located at the top of the RPV, includes electric heaters for pressurization and water spray for depressurization. The pressurizer has a relatively large steam volume of 105 m^3 at the rated operating condition. This large volume absorbs the changes in coolant volume that occur due to temperature changes in the primary system and mitigates pressure transients. Therefore, it contributes to the good controllability of the SPWR which has no control rods.

Reactor coolant pumps

The SPWR has only one main coolant pump (MCP) (Figure 6.4-3) which is located at the side of the RPV. With only one MCP installed, a MCP seizure accident was anticipated to be severe for the SPWR with respect to the DNBR (Departure from Nucleate Boiling Ratio). Preliminary calculations have shown, however, that a fuel burn-out will not occur because of the low core power density and the high natural coolant circulation rate capability due to low flow resistance and the high water head of the primary circuit.

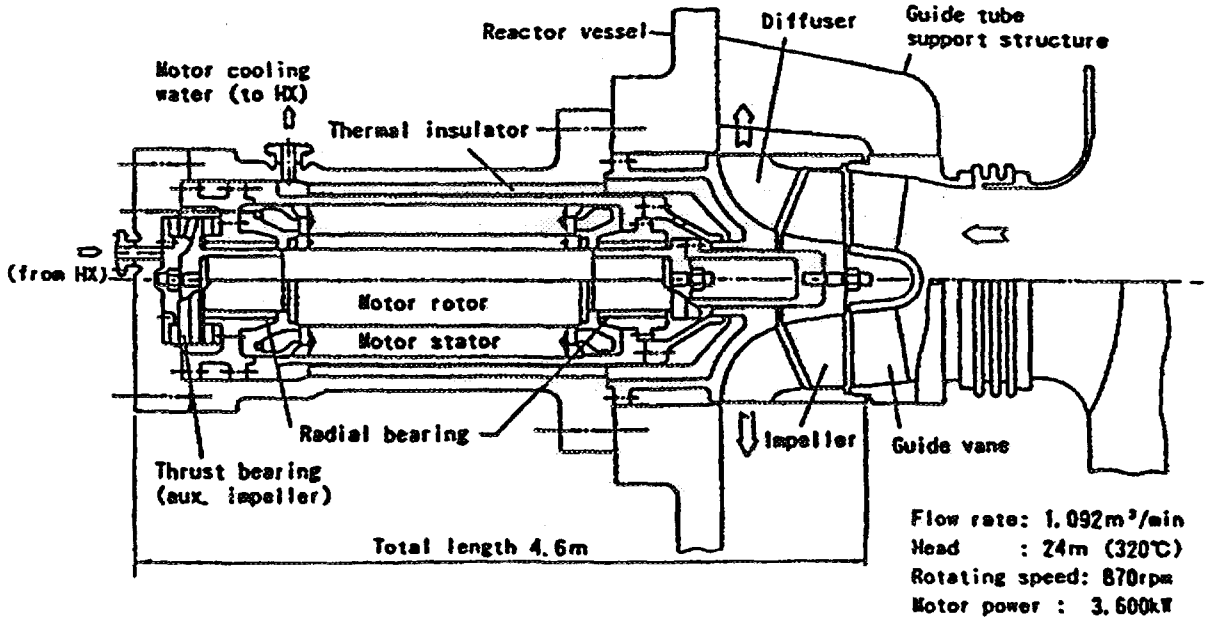


Figure 6.4-3 SPWR - Main reactor coolant pump (600 MWe)

Main coolant lines

The SPWR is an integral reactor concept and it has no main coolant lines; the primary coolant is circulated within the RPV.

6.4.2.5 Reactor auxiliary systems

The chemical and volume control system (CVCS) has similar tasks in the SPWR as in currently operating PWRs. It is an active non-safety grade system (1 train) and has a similar capability to that of the currently operating PWRs. The CVCS is used as a backup reactor shutdown system, if the passive reactor shutdown system (PRSS) were to fail.

As mentioned above, the RPV is surrounded by a water-filled containment vessel (CV). The RPV is thermally insulated and covered by a water-tight shell. The space between this water-tight shell and the RPV wall proper corresponds to the drywell of the pressure suppression type containments of currently operating BWRs. Taking into account that a pipe rupture may occur within this space, the shell is equipped with pressure relief valves.

The water-filled containment vessel has been found advantageous since it allows the reactor plant to be made very compact and it makes it easy to apply a passive decay heat removal system.

The containment vessel is made of steel plate of less than 38 mm in thickness; a thickness that does not require post-welding annealing. It is possible to design for even higher pressure and temperature using such a thin plate, because the containment vessel is compact. The design pressure and temperature of the containment vessel are 11 bar and 184°C, respectively. The large temperature difference between the containment vessel and the atmosphere contributes to making the Containment water cooling system (CWCS) compact.

6.4.2.6 Operating characteristics

Reactivity compensation for fuel burnup is achieved by changes of the boron concentration in the primary coolant, whereas changes in the reactor power level can be accomplished automatically as a result of changes in turbine load due to the strongly negative moderator temperature reactivity coefficient of the reactor core. Preliminary analyses have shown that the SPWR core has an excellent controllability, including reactor start-up operation, even though there are no control rods. Preliminary analyses show that the plant can easily accommodate 100%-50%-100% load changes simply by means of the moderator temperature reactivity feedback.

6.4.3 Description of turbine generator plant system

6.4.3.1 Turbine generator plant

The main steam system of the SPWR is similar to that of currently operating LWRs, and the turbine generator plant consists of one high pressure (HP) turbine and two low pressure (LP) turbines. A steam re-heater is installed between the HP and LP turbines.

6.4.3.2 Condensate and feedwater systems

The feedwater system consists of a main feedwater system and an auxiliary feedwater system.

The main feedwater system has two turbine driven pumps and a motor driven pump. The feedwater heaters are similar to those of currently operating LWRs. The auxiliary feedwater system (Section 6.4.6.2)has two types of pumps, turbine and motor driven pumps.

6.4.3.3 Auxiliary systems

No information provided.

6.4.4 Instrumentation and control systems

6.4.4.1 Design concepts including control room

The SPWR design is based on the assumption that the plant shall be controlled by an automatic control system based on artificial intelligence (AI) during normal operation with utilization of passive systems for mitigation of accidents.

Control scheme

Reactivity control by chemical shim is a well-established technology, proven in currently operating PWRs. Preliminary analysis shows that the controllability of the SPWR during operation, including the reactor start-up process, is excellent even though it has no control rods.

- Reactivity change due to fuel burn-up is slow and it can be easily compensated by control of boron concentration.
- Load changes can be followed-up automatically by means of the negative coolant temperature coefficient of reactivity.
- Preliminary analysis shows that the reactor power can follow a 50%-100% change in load. In this analysis, the average temperature of the primary coolant changes only 8°C even at the beginning of an equilibrium cycle (BOEC), at which the negative temperature coefficient is small; the core outlet temperature is almost constant, and the core inlet temperature changes 15°C. The Xe poisoning effect which occurs during slow power changes such as a daily load following is suppressed due to low core power density. The primary coolant temperature change due to load following is allowable because the steam pressure of the once-through helical coil type steam generator can be set independent of the temperature conditions of the primary coolant.
- Reactor start-up is easily performed by continuous boron dilution. The injection of pure water for boron dilution is performed by a pump which can be operated only when the MCP is operating. This design provides a fail-safe mechanism for avoiding the possibility of reactivity insertion accidents.
- A large core has a potential possibility of having spatial Xe oscillation. The control of power oscillation due to Xe is an important issue for the SPWR since it has no control rod. Analyses conducted on the 1800 MWt SPWR core in the manner commonly applied on PWRs, have confirmed that the SPWR core has a satisfactory stability against Xe oscillations.

6.4.4.2 Reactor protection system and other safety systems

No information provided.

6.4.5 Electrical systems

6.4.5.1 Operational power supply systems

No information provided.

6.4.5.2 Safety-related systems

No information provided.

6.4.6 Safety concept

6.4.6.1 Safety requirements and design philosophy

The design philosophy with respect to safety is based on the defence-in-depth strategy, applying prevention, protection and mitigation measures at several layers.

At the prevention level, a number of design measures are taken to reduce initiator frequency and to limit consequences. The primary circuit integration prevents large size LOCAs, and limits the consequences of a LOCA. The larger reactor pressure vessel implies that the distance between the core and the reactor pressure wall is larger which, in turn, yields reduced neutron fluence for the reactor vessel material; the risk of material degradation that can lead to vessel failure is practically excluded. A tough structure of the steam generator reduces probability of interfacing LOCAs, and the implementation of passive systems and battery backup power sources limit sensitivity to loss of power supply. The water-filled containment and the passive cooling system for its water reduce the frequency of total loss of heat sink. Increased design margins of the primary system reduces frequency of primary transients, and the large primary and small secondary coolant inventories reduce the impact of secondary side transients.

With respect to protection, typical examples related to LOCAs are: the large water inventory and the leakage limitations arising from the primary circuit integration; the automatic and passive water injection eliminating erroneous actions from operators; and the water-filled containment that will limit the break flows.

The negative moderator temperature reactivity coefficient and the passive reactor shutdown system protect against anticipated transients without scram (ATWS), and the low power density of the core, the large thermal inertia of the primary circuit and the large natural circulation capability protects against severe transients.

Design measures aimed at mitigation are delineated in Section 6.4.6.2.

The safety systems concept of the SPWR encompasses both passive and active systems Figures 6.4-4 and 6.4-5. Passive systems are entrusted to provide basic safety functions such as reactor shutdown, short-term decay heat removal (or safety injection), and long-term decay heat removal, while an active system is utilized for depressurization of the primary circuit. Other active and non-safety grade systems are used to prevent unnecessary challenges from threatening the safety, to minimize the time to terminate the abnormal conditions, and to protect the facilities as the property.

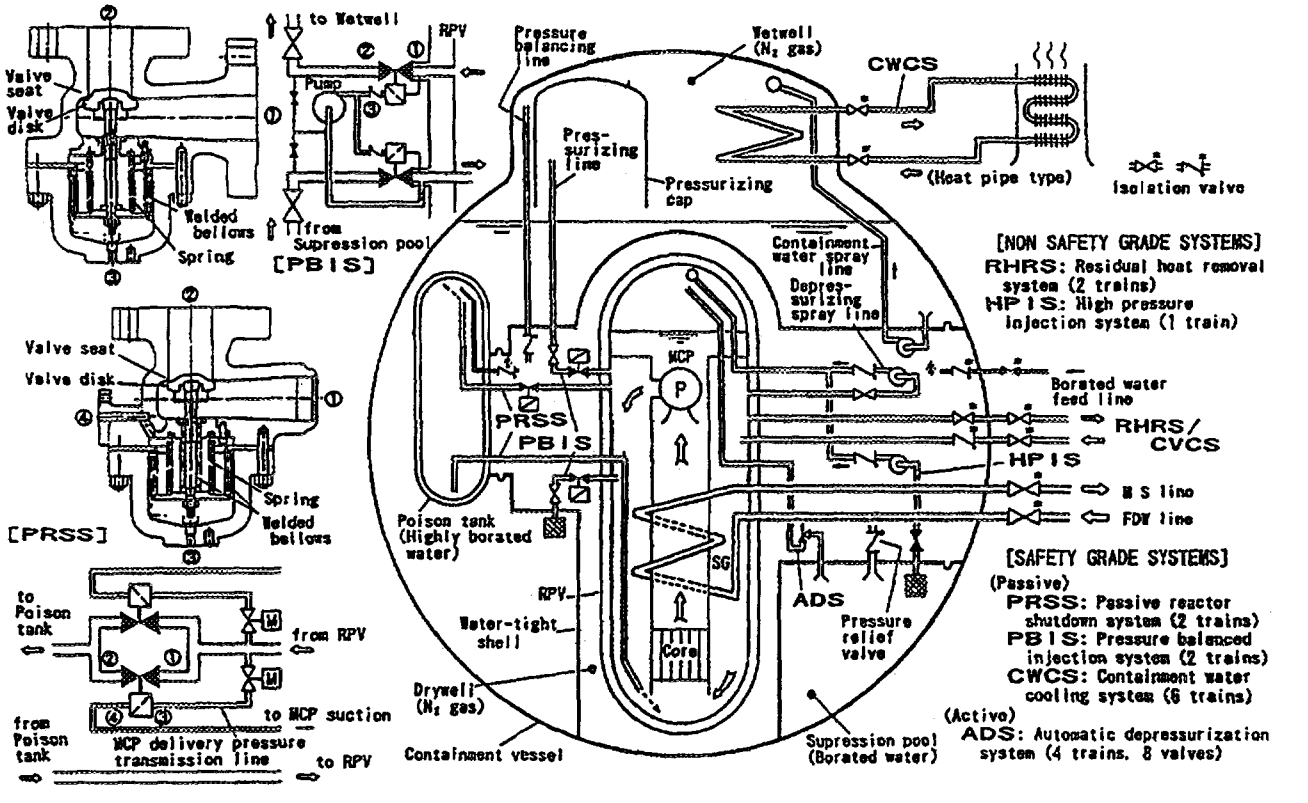


Figure 6.4-4 The SPWR safety systems concept

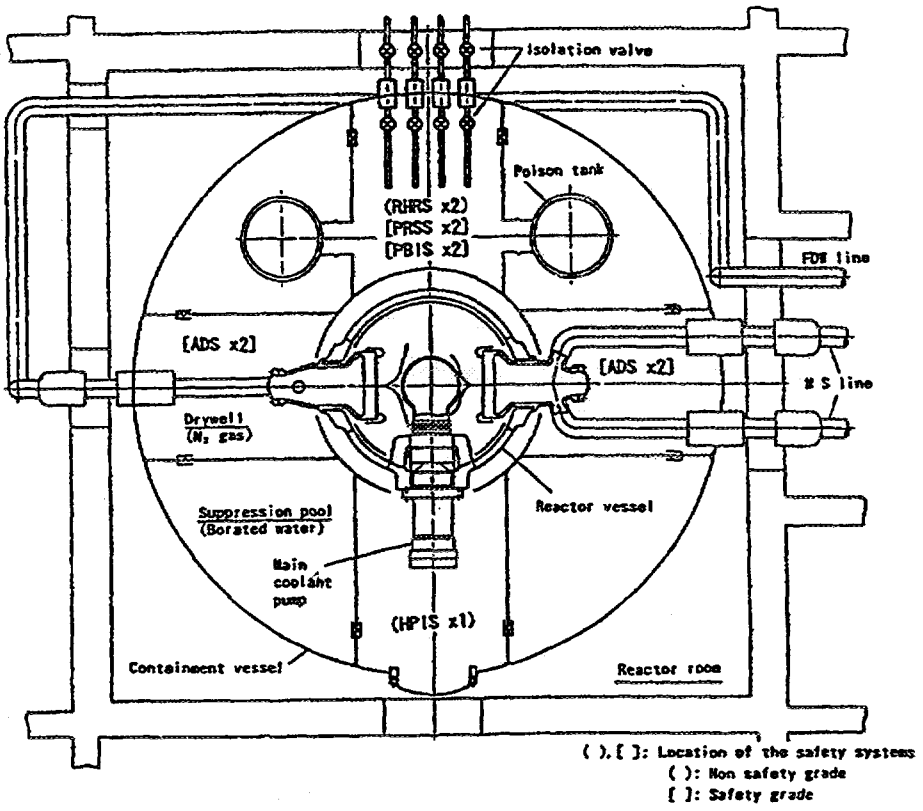


Figure 6.4-5 Plan view of the SPWR

Preliminary safety analyses have been performed on the earlier design (1100 MWt) with the thermal-hydraulic transient analysis code RETRAN-2 to understand the basic characteristics of the SPWR. Examples of the events analyzed are as follows: loss of coolant accident (LOCA), main coolant pump seizure, main steam line break, boron dilution, loss of main feedwater, loss of electrical load, and station blackout.

The results from the calculations, at beginning and end of equilibrium cycle (BOEC and EOEC, respectively) conditions, show that the SPWR is capable of maintaining the coolability of the fuel rods in the core under the above-mentioned accident situations, which are expected to cover the wide range of accidents to be considered in the licensing calculations, even if the functions of the reactor shutdown systems are not considered. A few examples of the dynamic behaviour are given in Figure 6.4-6.

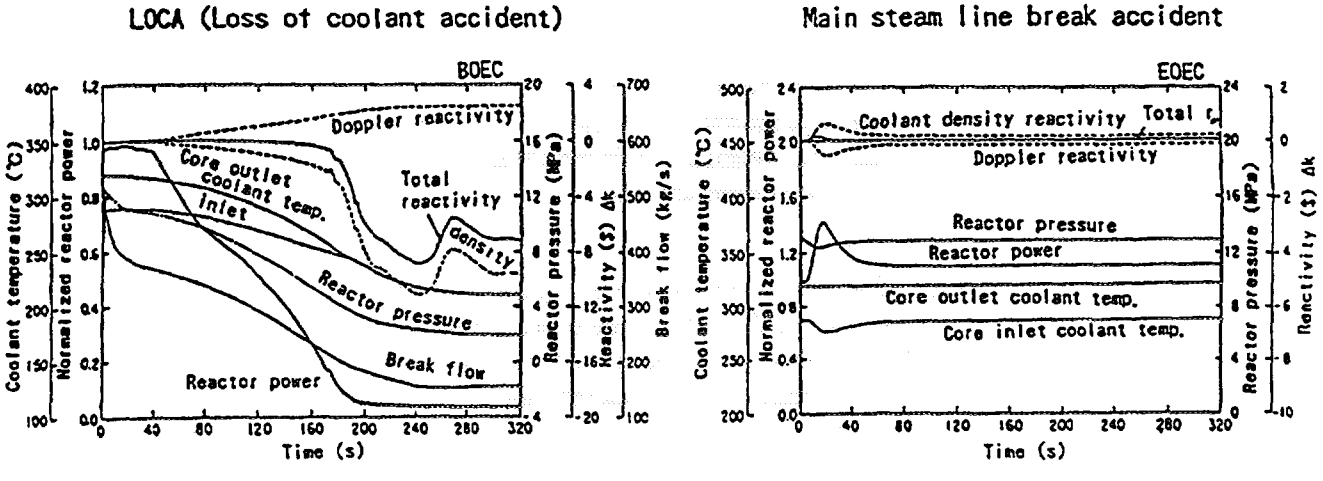


Figure 6.4-6 Examples on the dynamic behaviour of the SPWR

A probabilistic safety assessment has been performed to optimize the design of the SPWR, in particular with respect to the combination of passive and active safety systems. The preliminary results indicate that the estimated core melt frequency for all internal events is lower than 10^{-6}/reactor year.

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

The SPWR has a number of design-specific advantages and inherent safety characteristics, such as:

- the primary circuit integration in the RPV eliminates large break LOCAs; the largest pipe is 129 mm in diameter,
- a large primary coolant inventory allows safety injection systems enough time in LOCA situations,
- the always negative reactivity coefficient suppresses power increase under off-normal conditions, except over-cooling (similar to currently operating PWRs),
- lack of control rods and fail-safe design of clean water injection system eliminate possibility of large rapid reactivity insertion.

The SPWR employs passive systems to provide the basic safety functions of shutting down the reactor, and cooling the reactor core in both short and longer term by using the passive safety systems passive reactor shutdown system (PRSS), pressure balanced injection system (PBIS) and containment water cooling system (CWCS), supplemented by the active automatic depressurization system (ADS).

Passive reactor shutdown system

The passive reactor shutdown system (PRSS) is a poison injection system operating in natural circulation. It consists of two 100% trains, each with one poison tank, containing 60 m^3 of water with 12 000 ppm of boron, and with two hydraulic pressure valves. Under normal operation, the pressure of the poison tank is maintained at the same pressure as that of the primary system under normal operation. The borated water in the tank is cooled by the containment vessel water. The inside wall and the nozzle part of the poison tank are provided with liners for mitigation of the thermal stresses that will occur during poison injection, due to temperature changes.

The valve disk of the valves is kept in closed position by the delivery pressure of the main coolant pump (MCP) exceeding the spring force; the valves will open passively upon loss of the delivery pressure of the MCP. If the delivery pressure of the MCP is lost, the disk is pushed back to the open position by the spring force, letting borated water from the poison tanks into the primary loop. The boron content of the poison tanks is sufficient to maintain the reactor subcritical at cold shutdown condition.

The PRSS is also used as an active reactor shutdown system. While the MCP continues running, a scram signal can activate a motor-actuated valve in the MCP delivery pressure transmission line to close and thereby, to open the hydraulic pressure valves. In this case, the poison is injected in forced circulation by the MCP delivery pressure.

The time for the borated water in the poison tank to reach the core will be approximately 24 seconds in the natural circulation mode and approximately 8 seconds by forced circulation. The shutdown time of the SPWR is relatively long, but this is not regarded an adverse feature since the SPWR has no possibility of rapid reactivity insertion and system behaviours are mild.

Pressure balanced injection system

The Pressure balanced injection system (PBIS) is a passive system that injects containment vessel (CV) water (3200 m^3 of borated water at 2500 ppm boron) into the reactor pressure vessel in the event of a LOCA. The PBIS consists of two 100% trains, each of which has an injection line to lead containment vessel water to the RPV and a pressurizing line for equalizing the pressures in the RPV and the containment vessel. Steam from the RPV conveyed by the pressurizing lines flows into the pressurizing cap in the wetwell and replaces N_2 gas to rapidly equalize the pressure between the RPV and the containment vessel.

Each injection line or pressurizing line has a reliable hydraulic pressure valve actuated by the pressure difference between the RPV and the containment vessel to allow gravitational feed of water from the containment vessel to the RPV. The hydraulic pressure valve which is directly connected to the RPV is a check valve of a spring actuation type which does not open until reaching the design pressure difference (for example, 0.5 MPa) between the RPV and the containment vessel.

Containment water cooling system

The containment water cooling system (CWCS) is a heat pipe type passive heat removal system that consists of six 25% trains. The containment water coolers (evaporator of heat pipe) are installed in the upper part of the wetwell of the containment vessel. The heat transferred from the containment vessel water at the cooler is removed at the air coolers (condenser of heat pipe) by naturally circulated air. The capability of the containment water cooler under normal operation is low because the wetwell of the containment vessel is filled with N_2 gas. Therefore, the water in the containment vessel is sprayed to the coolers (the heat pipe evaporators) to achieve circulation of the heat medium (water) in order to maintain the normal operating condition of the CWCS.

Automatic depressurization system

The automatic depressurization system (ADS) is an active safety-grade system, taken from BWR technology, which has the functions of depressurizing the RPV to support the injection of containment vessel water by the Pressure balanced injection system (PBIS) in a high pressure accident situation and to prevent overpressurization of the RPV. The ADS consists of 4 trains, each of which has two relief valves. The objective of the active and non-safety grade systems of the SPWR is to prevent unnecessary challenges from threatening the safety, to minimize the time to terminate the abnormal conditions, and to protect the facilities as the property. The active systems draw upon the experience of similar well-proven designs used in currently operating BWR and PWR nuclear power plants.

High pressure injection system

The high pressure injection system (HPIS) is an active non-safety grade system (1 train) to inject containment vessel water into the RPV under LOCA conditions and to control the primary coolant inventory in transients, if necessary.

Residual heat removal system

The residual heat removal system (RHRS) is a non-safety grade system (2 trains). The RHRS circulates water in the RPV to a heat exchanger outside the containment vessel to remove decay heat under scheduled shutdown conditions as well as long term accident conditions. The heat exchanger is cooled by a component cooling system which transfers heat to a sea water system.

Chemical and volume control system

The chemical and volume control system (CVCS) is used as a reactor shutdown system for backup of the passive reactor shutdown system (PRSS). The CVCS is an active non-safety grade system (1 train) and has a similar capability to that of currently operating PWRs.

Auxiliary feedwater system

The Auxiliary feedwater system (AFWS) is a non-safety grade system to supply feedwater to the secondary side of the steam generator (SG) to remove heat from the core via the SG under a loss of main feedwater transient and so on. Two types of pumps, turbine driven and motor driven pumps are installed to achieve high functional reliability and diversity.

6.4.6.3 Severe accidents (beyond design basis accidents)

A severe accident leading to a core melt can not be ruled out, but specific design measures to cope with a molten core have not yet been incorporated in the SPWR concept. Given the water-filled containment and the passive cooling system for its water, it appears most likely that a molten core could be retained, and cooled, within the reactor vessel without penetrating it, by admitting the water mass available on outside the containment drywell, into it. A core catcher arrangement at the bottom of the cavity is under consideration, however.

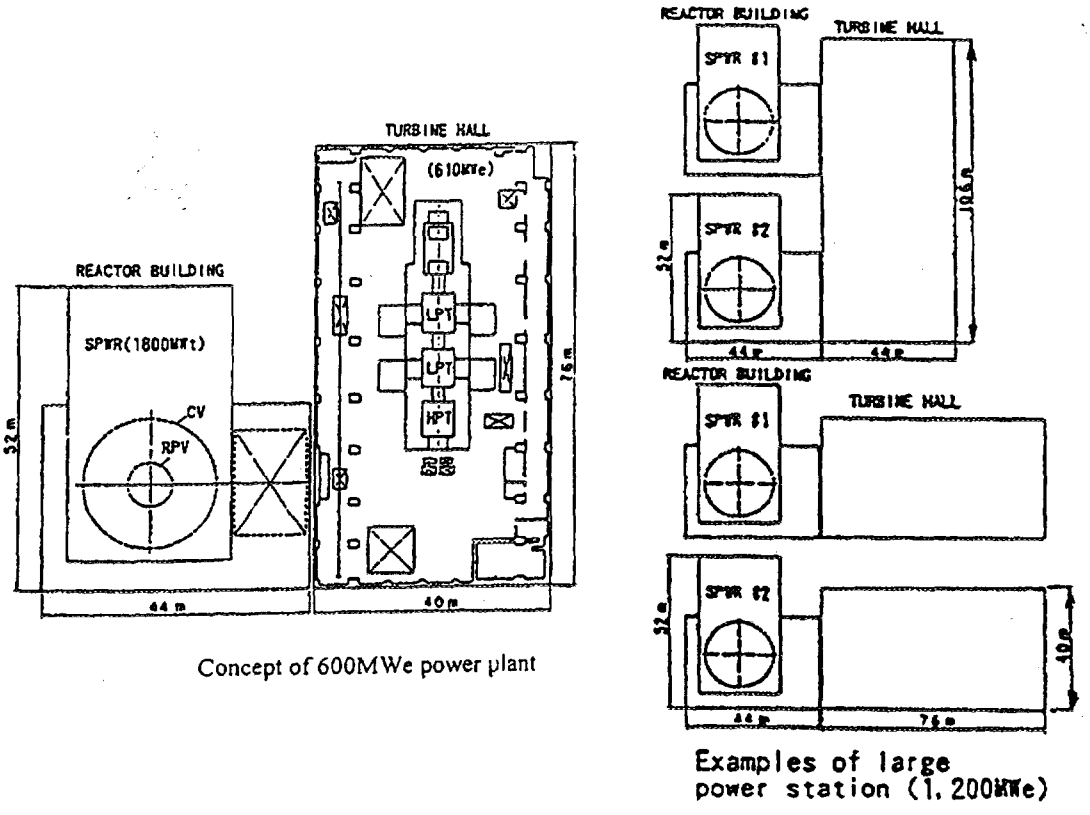


Figure 6.4-7 SPWR - Plant layout

6.4.7 Plant layout

6.4.7.1 Buildings and structures, including plot plan

The plant layout is shown in Figure 6.4-7. Examples of the layout for other plant power levels are also shown in the figure.

Seismic behaviour

The seismic design is considered much easier than for the currently operating LWRs because the reactor plant is simple and compact.

Design requirements

The SPWR will be designed to withstand earthquake, aircraft crash, explosion pressure wave, and internal hazards in a similar way as currently operating LWRs. In that context, physical separation of redundant trains of safety systems has been adopted, and accessibility, shielding, and ventilation are taken into consideration in the layout and installation work so that the plant can be operated at low occupational exposure of personnel and workers.

6.4.7.2 Reactor building

The concept of the reactor building is shown in Figure 6.4-8.

The reactor building is 53 m high and with a ground area of 52×44 m^2. The size of the reactor building is much smaller than that of currently operating LWRs due to the adoption of the small containment vessel and the simplification of plant systems.

The reactor pressure vessel, the primary circuit piping, valves and pumps can be inspected in the drywell, in a similar way as in currently operating BWRs. A preliminary study shows that only 36 days are necessary for a full inspection including refuelling.

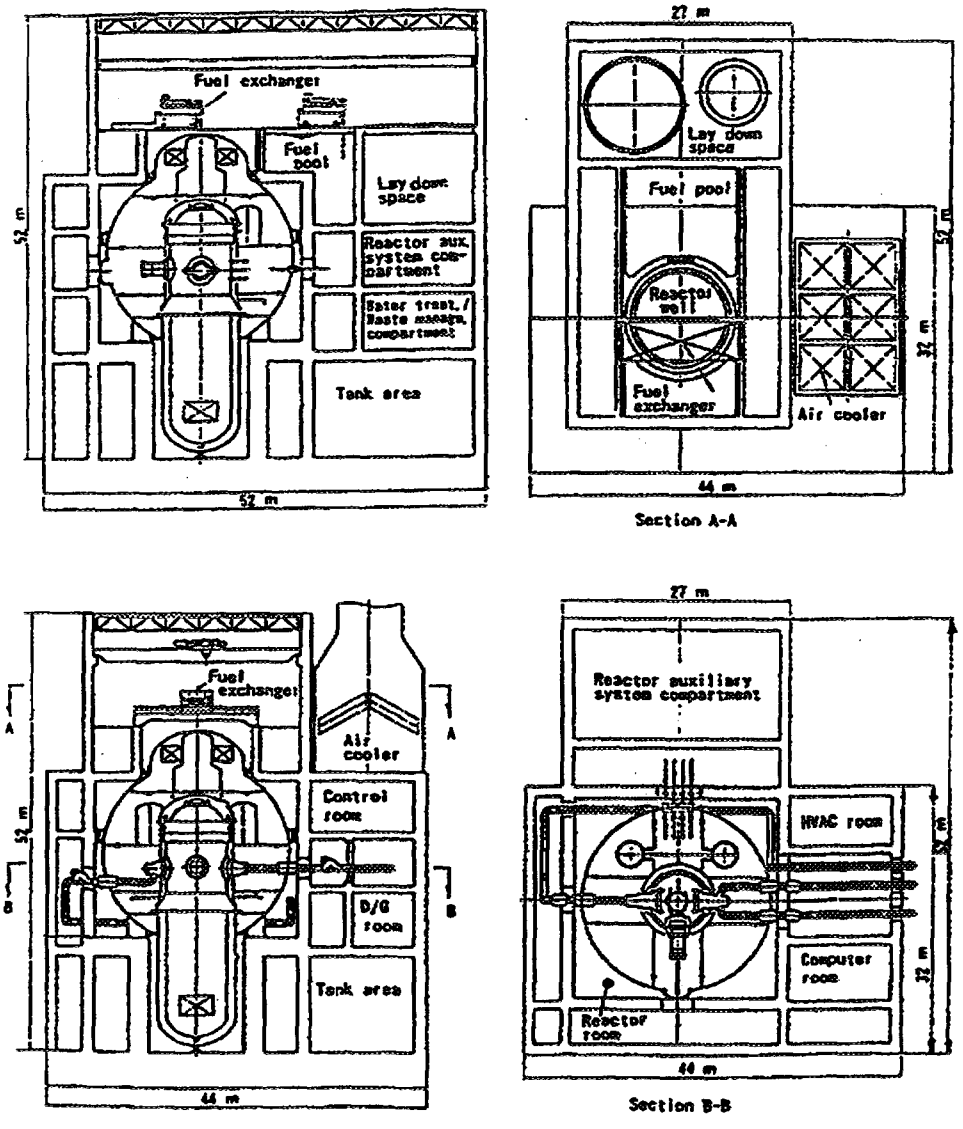


Figure 6.4-8 SPWR - Reactor building layout

6.4.7.3 Containment

The SPWR adopts a water-filled containment vessel. The RPV is covered with a water-tight shell with a mirror-type insulation of laminated thin stainless steel plates mounted inside the shell. The heat loss is estimated to be below 1 MW at operating conditions. The water is stored between the watertight shell and the containment vessel which gives a function corresponding to the wetwell of a pressure-suppression type containment as used by currently operating BWRs. The space between the water-tight shell and the RPV corresponds to the drywell of the pressure suppression type containment. Taking into account the possibility of a pipe rupture in this space, the shell is equipped with pressure relief valves. The advantages of applying the water-filled containment vessel are the compactness of the reactor plant and easiness of application of passive decay heat removal system.

The containment vessel is made of steel plate of less than 38 mm in thickness. Post-welding annealing is not required with this thickness. It is possible to design for higher pressure and temperature even with use of such a thin plate because the containment vessel is compact. The design pressure and temperature of the containment vessel are 1.1 MPa (11 bar) and 184°C, respectively. A large temperature difference between the containment vessel and the atmosphere contributes to making the containment water cooling system (CWCS) compact.

6.4.7.4 Turbine building

The concept of the turbine building is delineated in Figure 6.4-7. It has a width of 40 m and a depth of 76 m.

6.4.7.5 Other buildings

No information provided.

6.4.8 Technical data

General plant data		
Power plant output, gross	600	MWe
Power plant output, net		MWe
Reactor thermal output	1 800	MWt
Power plant efficiency, net		%
Cooling water temperature		°C
Nuclear steam supply system		
Number of coolant loops	1	
Primary circuit volume, including pressuriser	720	m^3
Steam flow rate at nominal conditions	905.5	kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	295/5.49	°C/MPa
Feedwater temperature/pressure	210/6.47	°C/MPa
Reactor coolant system		
Primary coolant flow rate	12 300	kg/s
Reactor operating pressure	13.5	MPa
Coolant inlet temperature, at RPV inlet	288	°C
Coolant outlet temperature, at RPV outlet	314	°C
Mean temperature rise across core	26	°C
Reactor core		
Active core height	2.4	m
Equivalent core diameter	3.83	m
Heat transfer surface in the core		m^2
Fuel inventory	74.8	t U
Average linear heat rate	12.6	kW/m
Average fuel power density	24.1	kW/kg U
Average core power density (volumetric)	65.1	kW/l
Thermal heat flux, F_q		kW/m^2
Enthalpy rise, F_H		

Fuel material	Sintered UO_2	
Fuel assembly total length	2 900	mm
Rod array	triangle, hexagonal	
Number of fuel assemblies	199	
Number of fuel rods/assembly	325 (30 with BA)	
Number of control rod guide tubes	-	
Number of spacers		
Enrichment (range) of first core		Wt%
Enrichment of reload fuel at equilibrium core	4.5 & 4.0	Wt%
Operating cycle length (fuel cycle length)	24	months
Average discharge burnup of fuel	48 000	MWd/t
Cladding tube material	Zircaloy	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly		kg
Active length of fuel rods	2 400	mm
Burnable absorber, strategy/material	Gd_2O_3 in fuel	
Number of control rods	0	
Absorber rods per control assembly		
Absorber material		
Drive mechanism		
Positioning rate		
Soluble neutron absorber	Natural boron	
Reactor pressure vessel		
Cylindrical shell inner diameter	6 600	mm
Wall thickness of cylindrical shell	285	mm
Total height	29 000	mm
Base material: cylindrical shell	JIS SFVQIA	
RPV head		
liner	stainless steel	
Design pressure/temperature	14.81/	MPa/°C
Transport weight (lower part), including head	1 534	t
RPV head		t

Steam generators

Type	Vertical, Once-through helical coil	
Number	1	
Heat transfer surface	17 000	m^2
Number of heat exchanger tubes		
Tube dimensions	15×2	mm
Maximum outer diameter	13 060	mm
Total height		mm
Transport weight	included in vessel above	
Shell and tube sheet material		
Tube material	Incoloy 800	

Reactor coolant pump

Type	Single-stage, centrifugal pump with canned motor	
Number	1	
Design pressure/temperature		MPa/°C
Design flow rate (at operating conditions)	12 000	kg/s
Pump head [at 320 °C]	24	m Wg
Power demand at coupling, cold/hot	-/3 600	kW
Pump casing material		
Pump speed	870	rpm

Pressuriser

Total volume	160	m^3
Steam volume: full power/zero power	105/77	m^3
Design pressure/temperature	/	MPa/°C
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter		mm
Total height		mm
Material		
Transport weight		t

Pressuriser relief tank

Not applicable

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Primary containment

Type	Water-filled, simple wall in steel (JIS SGV49,50)	
Overall form (spherical/cyl.)	spherical and cylindrical	
Dimensions (diameter/height)	22/38	m
Free volume	*75 000*	m^3
Design pressure/temperature (DBEs)	1.08/184	MPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate		vol%/day
Is secondary containment provided?	no	

Reactor auxiliary systems

Reactor water cleanup,	capacity	kg/s
	filter type	
Residual heat removal,	at high pressure	kg/s
	at low pressure	kg/s
Coolant injection,	at high pressure	kg/s
	at low pressure	kg/s

Power supply systems

Main transformer,	rated voltage	kV
	rated capacity	MVA
Plant transformers,	rated voltage	kV
	rated capacity	MVA
Start-up transformer	rated voltage	kV
	rated capacity	MVA
Medium voltage busbars (6 kV or 10 kV)		
Number of low voltage busbar systems		
Standby diesel generating units: number		
	rated power	MW
Number of diesel-backed busbar systems		
Voltage level of these		V ac
Number of DC distributions		

Voltage level of these		V dc
Number of battery-backed busbar systems		
Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	Steam turbine	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/ 2 LP	
Turbine speed	1 500 or 1 800	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature	5.30/292	MPa/°C

Generator

Type	3-phase, synchronous	
Rated power		MVA
Active power	610	MW
Voltage		kV
Frequency	50 or 60	Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type	
Number of tubes	
Heat transfer area	m^2
Cooling water flow rate	m^3/s
Cooling water temperature	°C
Condenser pressure	hPa

Condensate pumps

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

Condensate clean-up system

Full flow/part flow	
Filter type	

Feedwater tank

Volume	m^3
Pressure/temperature	MPa/°C

Feedwater pumps

Number	
Flow rate	kg/s
Pump head	
Feedwater temperature	°C
Pump speed	rpm

Condensate and feedwater heaters

Number of heating stages	
Redundancies	

6.4.9 P roject status and planned schedule

The design study has been performed by the Japan Atomic Energy Research Institute (JAERI) supported by the Science and Technology Agency (STA) since 1986.

The design concept of the SPWR has now been established. At the early stage, the design study was based on a poison tank installed in the RPV and with a reactor power of 1000 MWt (350 MWe). Thereafter, the present design was established, incorporating modifications such as relocation of the poison tank out of the RPV and raising the power to 1800 MWt (600 MWe).

A special team consisting of JAERI specialists independently performed a peer review of the SPWR design in 1992. The design team of the SPWR has improved the design in order to incorporate the advice made by the special team. The design study was completed in fiscal year 1993.

With respect to research and development work, the function of the hydraulic pressure valve for the PRSS has been successfully confirmed by a test with a half scale model. Verification tests will be performed on a number of other features. As a final step in the R & D programme, construction of a demonstration reactor is considered necessary, since there is no operating experience with reactors that lack control rods.

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6.5 VPBER-600, OKBM, RUSSIAN FEDERATION

6.5.1 Introduction

VPBER-600 is a medium power integral PWR placed in a guard vessel (GV). The main objective of the VPBER-600 design development was to create a medium power (600 MWe) reactor plant that is distinguished by a qualitatively high level of safety and by improved economic efficiency.

An enhanced safety reactor plant, the AST-500 heating reactor, was created in the early 1980s, and the principal solutions with respect to safety provisions of the AST, such as integral reactor layout, use of a guard vessel and use of passive safety systems, based on diverse operation principles with deep redundancy and self-actuation, were laid into the design basis when developing the VPBER-600 power reactor.

Protection of the plant personnel, the population, and the environment against radiation effects is the ultimate goal of the nuclear safety, and this was emphasized in the design objective for the VPBER-600. The prescribed doses of exposure, and the limits for release of radioactive substances to the environment, should not be exceeded under normal operation, anticipated operational occurrences, and in design-basis and beyond-design-basis accidents during 60 years of plant service life.

The nuclear power plant safety will be achieved by consistent implementation of the "defence-in-depth" principle based on the application of a system of barriers on the path of spreading ionizing radiation and radioactive substances into the environment, as well as of a system of engineered safeguards and organizational provisions for the protection of these barriers. A consistent implementation of the "defence-in-depth" principle implies:

- installation of successive physical barriers as noted above: fuel matrix, fuel cladding, primary circuit boundary, guard vessel and containment;
- taking into account all postulated initial events that can lead to a loss of efficiency of these barriers;
- determining for each postulated event design measures and actions of operating personnel, required to keep the integrity of the barriers mentioned and to mitigate the consequences of damaging such barriers;
- minimization of the probability of accidents resulting in the release of radioactive substances beyond the protective barriers;
- redundancy and diversity of systems, and physical separation of safety system trains.

A leak-tight primary system based on the use of canned reactor coolant pumps (RCPs), eliminating leaks during plant operation, and boron removal by ion-exchange filters, ensures a higher level of plant safety during normal operation as compared to that accomplished with traditional solutions.

As noted above, the plant design relies upon the use of a reactor with intrinsic self-protection properties, passive safety systems and devices, as well as upon self-actuated devices of "direct" principle of action. This limits unfavourable consequences of failures in the external systems, loss of power, plant personnel errors and of subversive actions.

The reactor self-protection features that limit the core power level, the rate of temperature rise in the reactor, and the rate of loss of coolant, are based on the following design parameters:

- reduced core power density;
- use of a core design with strong reactivity feedbacks (negative reactivity coefficients) and reduced need of boron concentration in the reactor coolant, and use of burnable poison;
- elimination of large diameter primary coolant pipelines, and use of flow restrictors;
- large volume of coolant above the core;
- high level of natural coolant circulation in the primary circuit provides effective emergency residual heat removal;
- reduced neutron fluence to the reactor pressure vessel (RPV) eliminates vessel embrittlement during its operation life.

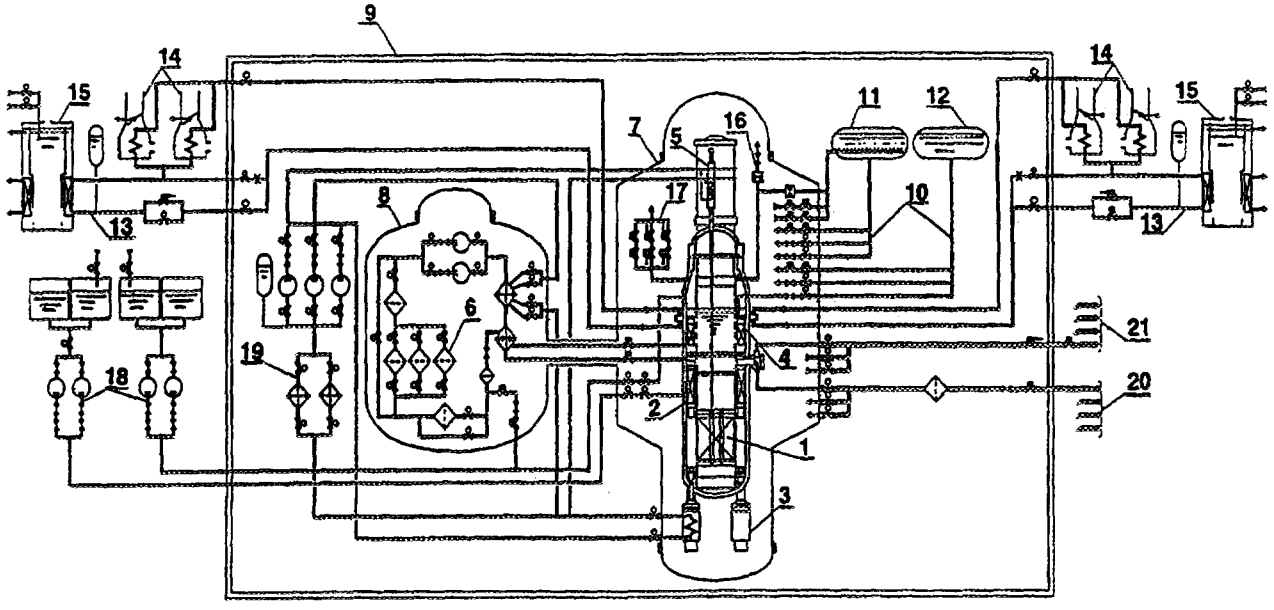


Figure 6.5-1 VPBER-600 - General flow sheet (Legend for numbering is given in Table 6.5-I)

In the design, passive systems are widely used which operate on the basis of natural processes and do not need external power supplies. Such systems are as follows:

- control rod drive mechanisms (CRDMs) provide control rod insertion into the core under gravity after de-energization of the drives, in response to signals of the reactor protection system, or immediately by direct action of the working medium;
- an emergency boron injection system will cause a boron solution to enter the reactor under gravity to trip the reactor;
- a passive emergency residual heat removal (ERHR) system provides cooling of the reactor for at least 3 days;
- the guard vessel will keep the core under coolant and provide the capability to cool the reactor down. In addition, the guard vessel acts as a reliable confinement of radioactive products after loss-of-coolant accidents;
- the containment protects the reactor plant against external impacts, and limits the release of radioactive products during beyond-design-basis accidents.

Table 6.5-I. Legend for Figures 6.5-1 - 6.5-6

No	Name	No	Name
1.	Reactor	15.	Water heat exchangers
2.	Steam generator	16.	Overpressure protection system
3.	Reactor coolant pump	17.	Reactor de-pressurization system
4.	Emergency RHR heat exchanger/condenser	18.	Primary coolant make-up system
5.	Control rod drive mechanisms (CRDM)	19.	Intermediate primary equipment cooling system
6.	Coolant purification and boron control system	20.	Feedwater
7.	Guard vessel of reactor	21.	Steam to consumers
8.	Guard vessel of purification system	22.	Removable parts of guard vessel
9.	Containment	23.	Guard vessel bottom
10.	Emergency boron injection system	24.	Lifting-transport machine
11.	Boron solution storage tank	25.	Refuelling machine
12.	Boron solution filled hydroaccumulators	26.	Reactor core
13.	Emergency RHR system	27.	Reactor pressure vessel
14.	Air heat exchangers	28.	Reactor closure head

The reliability of the safety systems is improved by the use of self-actuated devices which act upon the variation of working medium parameters, such as primary system pressure, guard vessel pressure, or coolant level in the reactor.

6.5.2 Description of the nuclear systems

A schematic diagram of the major reactor plant systems is shown in Figure 6.5-1, and the basic arrangement of the reactor system is depicted in Figure 6.5-2.

A legend for the name of systems and components shown in these as well as other figures is provided in Table 6.5-I.

6.5.2.1 Primary circuit and its main characteristics

The integral reactor is characterized by the arrangement in a common pressure vessel of the core with its control and protection system's members, the heat exchanger surfaces of the steam generators, heat exchanger-condensers of the emergency heat removal system and the steam-gas pressurizer.

The function of the latter is performed by the upper plenum above the coolant surface in the reactor vessel (Figure 6.5-3).

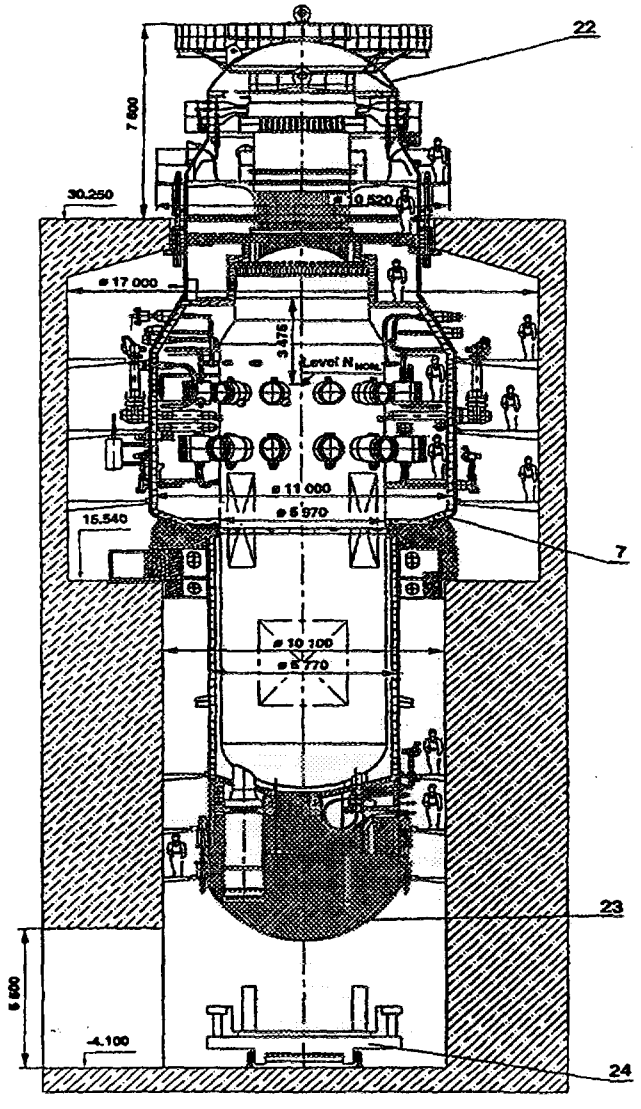


Figure 6.5-2 VPBER-600 - Reactor in guard vessel (legend is given in Table 6.5-I)

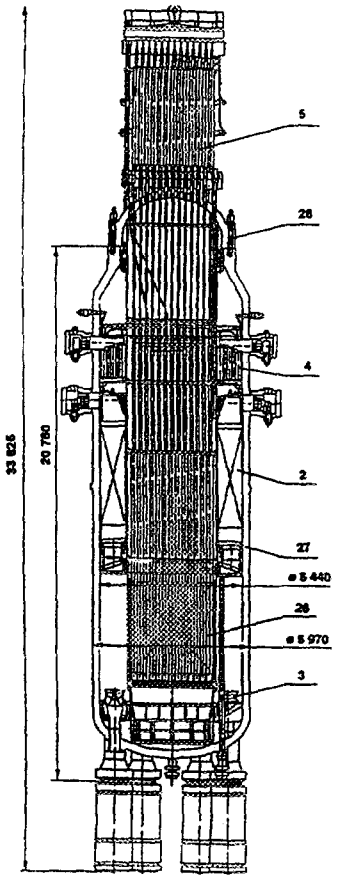


Figure 6.5-3 VPBER-600 - Reactor assembly (legend in Table 6.5-I)

The primary circuit includes the reactor coolant flow path and pressurizing system enclosed in the reactor pressure vessel, as well as the purification and boron reactivity control system connected to the reactor when the plant is in operation. The primary circuit provides heat removal from the core and transfer of the heat to the secondary coolant in twelve steam generator sections.

The integral reactor design eliminates the main coolant lines external to the reactor vessel and excludes, in essence, the classes of large and medium LOCAs due to primary circuit pipeline ruptures. The total volume of the integral reactor together with the cover is 340 m^3 of which 260 m^3 belong to coolant and 80 m^3 to steam-gas pressurizer. Normal core heat removal is effected in the forced circulation mode by six canned reactor coolant pumps. The pumps are built into the reactor bottom.

The heat-exchange surfaces of the steam generators are arranged above the core, in the annular gap between the reactor pressure vessel and the in-vessel barrel (Figure 6.5-4). The steam generator is of the once-through type, consisting of 12 independent sections. At the core inlet there is a pressure chamber ensuring uniform distribution of the coolant flow through the fuel assemblies of the core.

The simplicity of the circulation circuit assures a high degree of natural circulation flow, and the capability for a reliable cooling of the core by natural coolant circulation in all emergency situations, including the steam-condensation mode after loss-of-coolant accidents.

One of the features of the integral layout is a large water gap between the core and the reactor vessel serving as a radiation protection. As a result, the neutron fluence is less than 10^{17}n/cm^2 which eliminates the problem of reactor vessel metal degradation under irradiation.

Above the steam generators, heat exchangers submerged in the coolant provide for emergency removal of reactor residual heat. In the event of loss of primary circuit integrity the heat exchangers will operate as condensers.

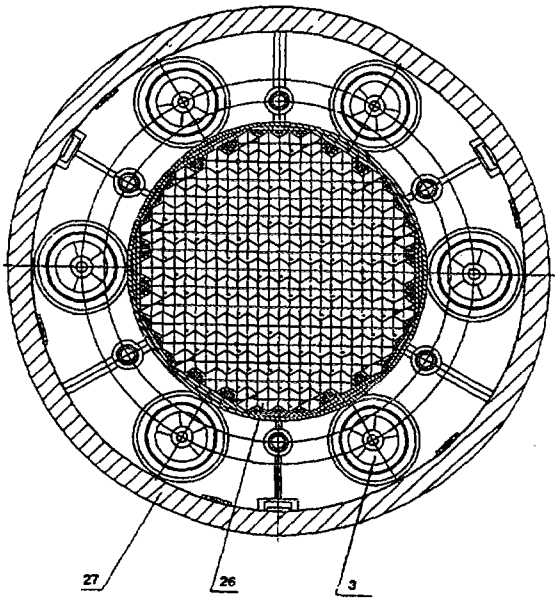


Figure 6.5-4 VPBER-600 - Reactor cross-section (Table 6.5-I)

The upper block of the reactor consists of the reactor cover, the in-vessel barrel structure and cables. The electromechanical control rod drives mechanisms (CRDMs) are installed on the reactor cover. They move the working members of the control and protection system in the power regulation mode, and release them under emergency protection conditions.

A highly effective mechanical system for reactivity compensation is used. The working members of the control and protection system (CPS) are available in the majority of the core fuel assemblies. The mechanical system can ensure a subcritical state of a cold and fresh core without boric acid addition to the moderator. Compensation of the reactivity margin for fuel burnup is provided by combined action of the control system members, boric acid in the coolant and self-shielded burnable poison.

The core consists of hexagonal fuel assemblies with fuel rods similar to those used in WWER reactors. The core consists of hexagonal fuel assemblies with fuel rods similar to those used in WWER reactors. The core allows realization of various fuel cycles distinguished in the number of reloadings per core lifetime. The average fuel burnup is 52 000 MWd/tU. The reactor refuelling interval is two years.

The ionization chambers for monitoring of the core are suspended at core level in a space between the in-vessel barrel and the reactor pressure vessel.

The main parameters of the reactor primary circuit are in short: the thermal power rating is 1800 MW; the operating pressure is 15.7 MPa; the coolant temperature at the core inlet and outlet is 294.4 and 325°C, respectively; and the nominal coolant flow rate amounts to 10 140 kg/s.

The reactor and the purification/boron control system connected to it during the plant operation are placed into a guard vessel (GV).

6.5.2.2 Reactor core and fuel design

The core consists of 151 hexagonal fuel assemblies (FAs) (size across flats is 234 mm) with fuel rods and fuel lattice parameters analogous to those in WWER-1000. Each fuel assembly contains boron carbide rods which are united to a cluster which forms a control device. The control devices of 135 fuel assemblies are connected to drives of the electromechanical control and protection system (CPS). The core height is 3.53 m, its equivalent diameter is 3.05 m and the average power density is 69.4 kW/l.

The moderate power density of the core has made it possible to reduce the content of boric acid in the coolant compared to that of the WWERs, to dispense with operational control of boric acid concentration for power changes (e.g., for load following) as well as for startup and shutting down of the reactor. Slow compensation of reactivity for fuel burnup is provided by ion-exchange filters operating in the leak-tight closed purification and reactivity control system.

The reduced content of boric acid ensures negative values of void and coolant temperature reactivity coefficients over the entire range of temperature variation. This allows to provide reactor self-shutdown at loss of primary circuit integrity and power self-limitation in emergencies with rise of power and temperature.

The core power density and increased margins with regard to the critical heat flux ensure the thermal-technical reliability of the core under normal operation and accident conditions.

6.5.2.3 Fuel handling and transfer systems

Handling of in-vessel structures and fuel during refuelling is made by the universal machine, developed by OKBM.

The refuelling machine moves on rails laid on the reactor service room floor and covers the area where the reactor pit, spent fuel storage pool and equipment storage pits are arranged. When manipulating with fuel, the refuelling machine is used for reshuffling fuel assemblies inside the core, transferring them between the core and spent fuel storage pool and inside the pool.

All manipulations with fuel assemblies beyond the reactor boundary are performed using a special water-filled transfer tube.

6.5.2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel is a factory-built welded vessel of 20.15 m height and 5.97 m outer diameter with elliptical bottom.

The arrangement of a circulation circuit inside the reactor vessel eliminates large diameter recirculation pipelines and hence the accidents with large and medium primary coolant leaks. All pipelines are connected to the reactor in its upper part and the vessel nozzles are provided with flow restrictors with equivalent diameters of 50 mm and 32 mm in the steam-gas and water volumes, respectively.

The arrangement with the steam generator inside the reactor pressure vessel structurally results in an enlarged water gap between the core and the vessel wall which reduces the neutron fluence down to a value of 7 10^{16}n/cm^2, for a reactor vessel service life of 60 years.

Reactor internals

No information provided.

Steam generators

The heat exchange surface of the steam generator (SG) is arranged above the core in the annular gap between the reactor pressure vessel and the in-vessel barrel. The steam generator is of once-through, cassette-type. The maximum possible inter-circuit leak is equivalent to an orifice of 24 mm diameter for feedwater and of 40 mm diameter for steam. The SG cassettes are combined in twelve independent sections with individual supply of feedwater and removal of steam out of the reactor and guard vessel.

Pressurizer

The pressurizing system is intended for creating and maintaining the primary circuit preset pressure at variation of the coolant volume during the reactor operation. The pressurizer is formed by a steam-gas plenum under the reactor cover. For initial filling of the pressurizer, nitrogen is used. The additional

pressurization is achieved by steam evaporation from the primary coolant surface. The partial pressure of nitrogen above the coolant level is 3.9 MPa.

Reactor coolant pumps

In the bottom of the reactor pressure vessel there are six built-in reactor coolant motor pumps. The pump flow rate is 8250 m^3/hr (1690 kg/s); the delivery head is 5.14 MPa. The pump drive is a canned electric motor of 2870 kW power at a speed of 1500 rpm.

The design of the pump and of the nozzle in which the pump is mounted, is such that at a guillotine rupture or a breach in any possible place of the nozzle the cross section for coolant outflow from the reactor will be equivalent to a 17.5 mm diameter orifice.

Main coolant lines

The integral design implies that the coolant is circulated inside the reactor pressure vessel and there are no external main coolant lines.

6.5.2.5 Reactor auxiliary systems

Chemical and volume control system

The purification and boron reactivity control system serves to maintain required primary coolant quality during operation and for periodic removal of excessive boron (15-20 times for core life) to compensate for fuel burnup. The system includes a recuperator, cooler, pumps and ion-exchangers.

Guard vessel (GV)

The guard vessel is a passive protective and confinement device ensuring safety at loss-of-coolant-accidents (LOCAs), by fulfilling the following functions:

- keeping the core covered by coolant;
- confining radioactive products which escape beyond the reactor boundary in case of coolant leaks.

The guard vessel is a factory-built steel vessel consisting of several parts assembled together at the site.

6.5.2.6 Operating characteristics

In accordance with the requirements on manoeuvrability of NPPs for Russian nuclear power stations to be operated in the semi-peak regime, VPBER-600 is capable of operating continuously at power levels between 30 and 100% of rated power.

Transition of the reactor plant from one power level to another is accomplished through changes in the flow rate of feedwater to the steam generators provided by a feedwater control complex. Xenon reactivity effects in transients are compensated by movements of the control rods only, without changes of boric acid content in a moderator. The feedwater control complex has a linear characteristic and provides changing of feedwater flow with the rate of 0.5%/min to 10%/s. In the power range, the "gray" absorber rods are moved only, while in the lower power range the "black" control rods are used.

During load variation the main steam pressure is kept constant at 5.9 MPa, and the pressure in the primary system does not change very much; it varies between 15.4 and 15.7 MPa. Other parameters follow the quasi-static characteristics (according to the part load diagram). The reactor plant provides daily load variation according to one of the following algorithms:

- 17 h at 100% of rated power - 2 h power reduction - 3 h at 30% - 2 h power rise, or.
- 14 h at 100% - 2 h reduction - 6 h at 30% - 2 h rise.

A weekend load variation mode with power reduction down to the level of 15 to 17% of rated power is also provided to follow a load drop during a weekend.

The maximum allowable rate of load variation within a 70% range of power regulation is 2.5%/min, within a 40% range 10%/min and within a 10% range it is 1.0%/s.

The VPBER-600 power unit provides power reduction modes keeping the plant running without actuation of the reactor protection system even if failures in the steam turbine plant occurs, such as: trip of one condensate or feedwater pump, electrical load rejection, de-energization of the generator, trip of the turbine, isolation of a steam line to the turbine plant.

6.5.3 Description of turbine generator plant system

6.5.3.1 Turbine generator plant

In the VPBER-600 nuclear power plant either a condensing turbine of K-600-5.9/500 type for power plant or an extraction turbine of T-600-5.9/500 type for a co-generation plant is used.

The turbine with intermediate steam separation and one-stage superheating without controlled bleeds is intended for direct driving of alternating current generator of T3B type with water cooling, of 630 MW power and 3000 rpm speed.

In the design with condensing turbine, the maximum electric power is 640 MW, the heat output is up to 215 GCal/h, (250 MW).

In the design with steam-extraction turbine the heat output is up to 645 GCal/h (with reduction of the electric output down to 430 MW).

The turbine consists of one cylinder in the high-medium pressure section plus two cylinders in the low pressure section with seven uncontrolled bleeds to four low pressure reheaters, a deaerator and two high pressure reheaters.

6.5.3.2 Condensate and feedwater systems

The steam-turbine plant includes also two condensers, three condensate pumps (one as back-up), three electric feedwater pumps; condensate and reheaters, separate water purification system and a heating facility.

6.5.3.3 Auxiliary systems

No information provided.

6.5.4 Instrumentation and control systems

6.5.4.1 Design concepts including control room

The plant operation is controlled by the integrated automated control system on the basis of multiple redundant control computing devices with automatic diagnostic of software-hardware. The technical capabilities of the computing devices allow to perform continuous and periodical diagnoses of the state of the most important elements and the reactor plant as a whole by all presently used methods which considerably reduces the probability of sudden failures.

The automated system allows to solve the problems of the plant automatic control, monitoring and diagnosing at normal operation and reliable actuation of safeguards, as well as to diagnose the plant actual state if an accident occurs.

The automated control system has a decentralized hierarchical structure (Figure 6.5-5.); safety actuation systems are separated from normal operation systems to such extent that failures in the latter systems do not jeopardize the functional capability of the former.

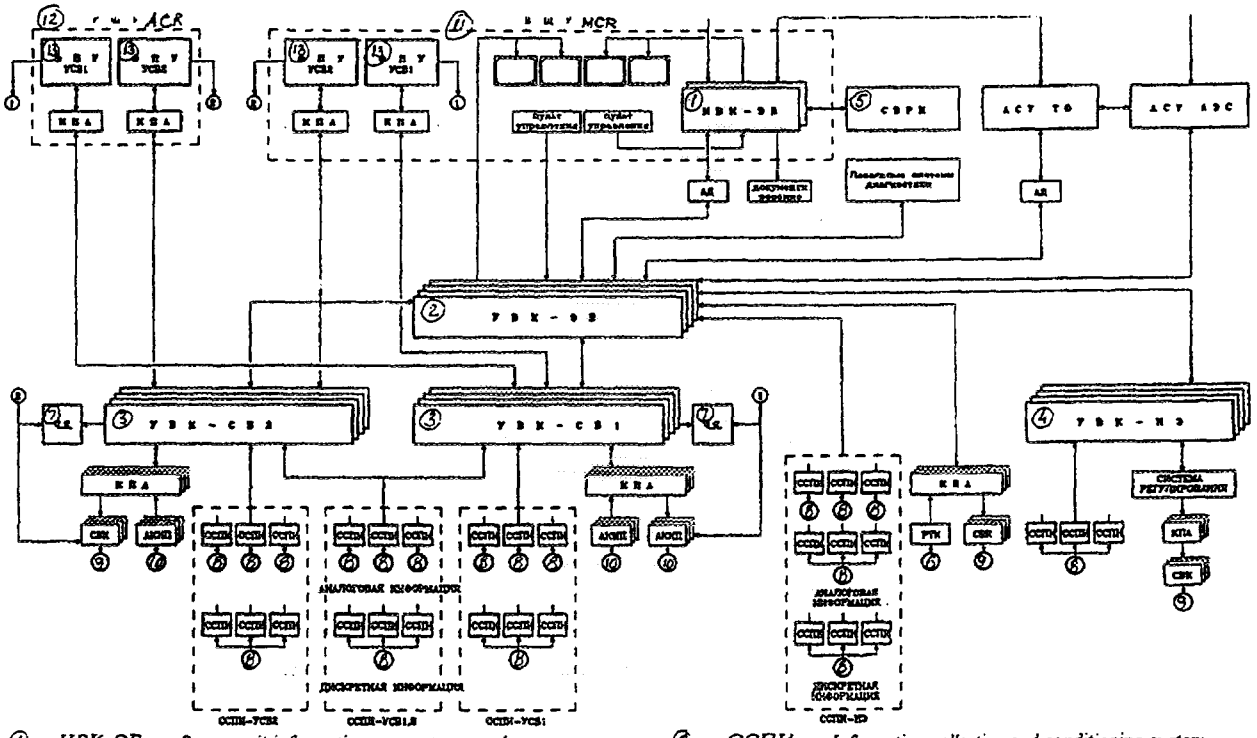


Figure 6.5-5 VPBER-600 - Instrumentation and control systems structure

Devices of other principle of operation are also used in order to enhance the protection of the reactor plant. These devices are actuated by direct action of working medium. Pressure in the reactor and in the GV and coolant level in the reactor are used as activating parameters of safeguards systems. The complex of self-actuated features provide for reactor shutdown, actuation of the emergency heat removal system, closing of isolation valves on the steam generator in all anticipated accidents.

6.5.4.2 Reactor protection system and other safety systems

The fulfilment of the reactor protection function and initiation of the safety systems are provided by the safety actuation systems. These systems consist of two sets of automatic means which are divided electrically and segregated in order to exclude any influence of failures in one of the sets to the other, as well as to prevent simultaneous failure to function of the sets under external impacts.

Each set comprises three control trains. Protection action signals are generated in the sets in a "2-out-of-3" logic, whereas a 1-out-of-2 logic is used between the sets.

The reactor control and protection systems use automatic means with digital data processing and fibre optical links for data transmission.

6.5.5 Electrical systems

6.5.5.1 Operational power supply systems

The basic arrangement of the electrical systems in the VPBER-600 unit is shown in the diagram on Figure 6.5-6.

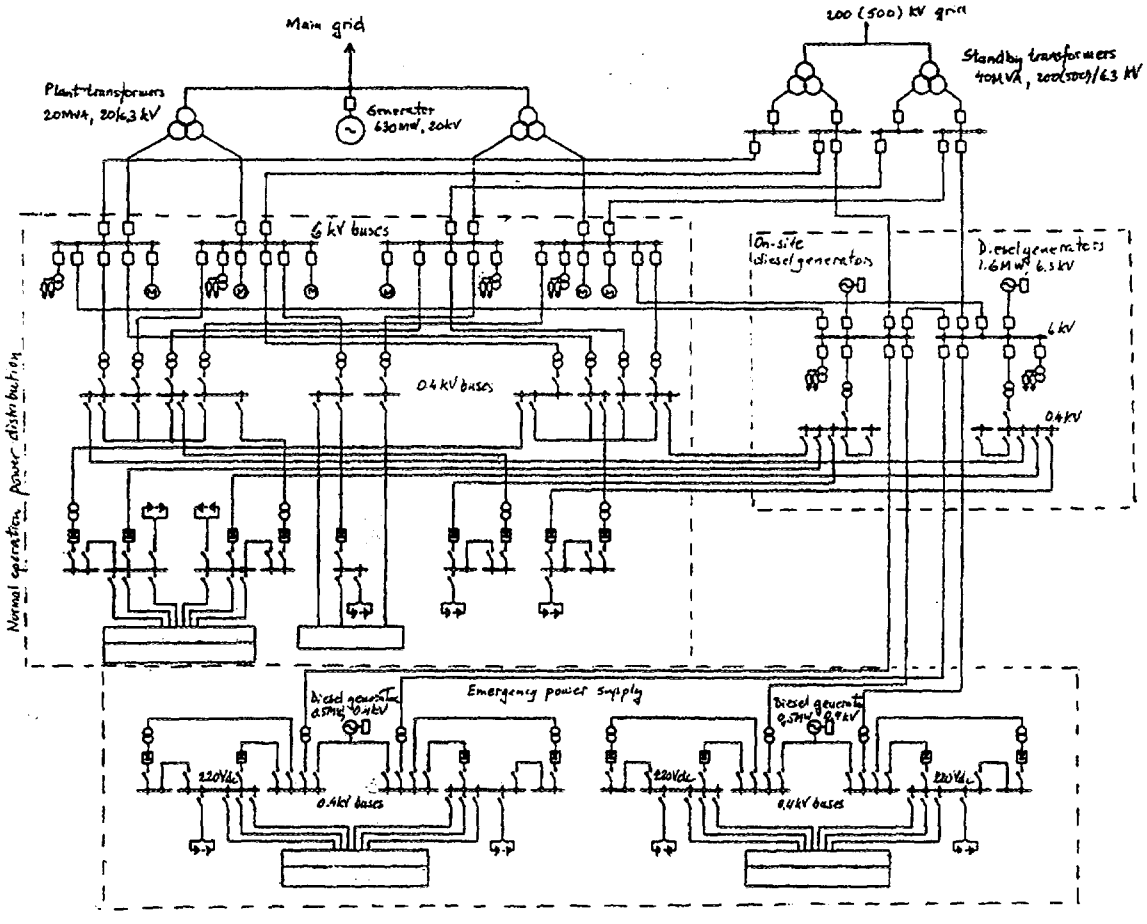


Figure 6.5-6 VPBER-600 - Electrical systems structure

Power supply to all the power consumers of the unit is provided from the plant power and stand-by transformers. The plant power transformers are connected to the main generator circuit (generator step-up transformer) from a low voltage side, while the stand-by transformers are connected to busbars of high voltage switchgears which are connected by transmission lines to an off-site power grid. There are two 20/6.3 kV plant power and two 220(500)/6.3 kV stand-by transformers.

To ensure availability of essential equipment items in the loss-of-power event (loss of the turbo-generator and an off-site grid), two 1.6 MW diesel generator sets with automatic startup are provided as well as battery sets for power supply to the automatic control systems during startup of the diesel generators.

6.5.5.2 Safety-related systems

An emergency power supply system is envisaged in a structure of the safety systems. The system consists of two electrically and spatially separated trains, which preserve their functional capability under design basis earthquake conditions. Each train comprises two battery sets that provide power supply to the actuation safety systems and one 0.5 MW diesel generator. Taking account of a priority given in the reactor plant design to passive safety features, automatic startup of the diesel generator has not been envisaged. The capacity of the battery sets is sufficient for 72 hours of operation; the fuel reserve for the diesel generator sets will last for 7 days.

Power delivery to the emergency power supply system trains is provided from:

- the plant power and stand-by transformers - under the normal power supply conditions;
- the on-site diesel generator sets - under loss-of-power conditions;
- the battery sets and from the remotely actuated diesel generators of the system - under loss-of-power conditions and failure of the on-site diesel generators to start.

6.5.6 Safety concept

6.5.6.1 Safety requirements and design philosophy

The reactor plant enhanced safety is based on the realization of engineering decisions which are aimed at solving two main safety tasks:

- enhancement of self-protection properties intrinsic to the integral PWR;
- development of functional and physical in-depth protection.

The functional protection includes the means for the reactor shutdown, residual heat removal and confining radioactive products.

Reactor shutdown features

The reactor trip in emergency is done by simultaneous insertion of the control devices into the core by gravity following de-energization of the drives, which is effected both by signals from the automatic control system, as well as by direct action of the working medium: increased pressure in the reactor or or in the guard vessel.

In case of failure to actuate the electromechanical protection system reactor shutdown is accomplished by the emergency boron injection system. Activation of the system is possible by opening valves in the pipelines connecting the system to the reactor or by rupture of a membrane with simultaneous opening of check valves in the drain line by direct action of the pressure in the reactor. The boron solution is supplied by gravity due to the location of the system tanks above the reactor.

Each of the systems ensures the reactor shutdown and is capable to keep the cold dispoisoned reactor in a subcritical state.

Emergency heat removal means

The emergency heat removal system ensures the removal of residual heat from the core should the reactor cooling through the steam turbine plant become impossible.

The system includes two blocks of heat exchangers forming a four-train passive heat removal system, removing heat into water storage tanks.

The system operates at natural circulation of coolant in all the circuits. The pressure in the intermediate circuit of the system is higher than those in the reactor and thereby a barrier against radioactive products propagation is provided.

The system of passive heat removal is actuated, if an emergency arises, by opening the valves on the pipelines for discharge of water from the heat exchangers in response to signals from the automatic control system, as well as directly by the action of the working medium: to signals of pressure or coolant level variation in the reactor.

The heat being removed from the reactor via built-in heat exchanger-condensers is transferred to water storage tanks through which cooling water is circulated. In case of loss of cooling water flow the removal of heat is effected by evaporation of water from the tanks. The steam is discharged into the atmosphere. It is possible to use air heat exchangers capable to remove the reactor residual heat for unlimited time after cooling water evaporation from the water storage tanks.

Radioactive products confinement means

The original engineering solutions concerning retention of radioactive products are as follows:

- leak-tight integral reactor with outflow restrictors in the penetrations of auxiliary systems through the reactor vessel (equivalent diameter 50 mm in the steam-gas plenum, 32 mm in the water volume and 17.5 mm in the pump nozzle);

- cassette-type steam generator with structural limitation of possible inter-circuit leak (equivalent diameter 40 mm for steam, and 24 mm for feedwater);
- factory-built guard vessel which is designed for the pressure that can develop upon loss of primary circuit integrity;
- steam generator isolation from the steam turbine plant at intercircuit leakage is provided by three isolation valves, two of which are closed both automatically and remotely by operator to the signal of lowering water level in the reactor, the third one is closed remotely by operator only;
- steam generator which is designed for primary circuit pressure up to the isolating means;
- leak-tight intermediate circuit of the emergency heat removal system with a pressure barrier in the direction to the reactor;
- leak-tight intermediate circuit for cooling of the reactor plant equipment which is designed for primary circuit pressure.

Complex of systems for ensuring safety by passive means

The reactor plant safety is ensured without power supply and personnel intervention for not less than 72 hours at all possible disturbances and accidents (positive reactivity insertion, loss of heat removal, primary circuit depressurization) along with the reactor inherent safety features also by using the complex of interconnected passive safety means (systems and devices).

This complex includes:

- guard vessel;
- emergency heat removal system;
- emergency boron injection system.

Upon positive reactivity addition or loss of reactor heat removal following the reactor trip by the electromechanical protection system or the emergency boron injection system, the core residual heat removal is effected by the passive emergency heat removal system. The amount of water in the tanks of the system ensures reactor cooling for at least 72 hours (seven days with two tanks and three days with one tank available).

In case of a LOCA the leakage of primary coolant from the reactor terminates at relatively high pressure in the system "reactor-guard vessel" (up to 3.6 MPa) due to the small volume of the guard vessel. The temperature developing in this system is sufficiently high, which enables to remove the residual heat via the emergency heat removal system. The amount of water in the reactor and in the emergency boron injection system which is connected to the reactor through check valves is sufficient to keep the core under water for 72 hours.

Deterministic design basis

All technically possible failures of the plant equipment and systems have been considered in the course of VPBER-600 safety analysis. The most significant failures which are initiators of emergency situations and of accidents have been analysed taking into account probable coincident failures in the accidents. Analyses have been carried out for different kinds of initial events: addition of positive reactivity (spontaneous removal of control rods, breaks of steam lines etc.), deterioration of heat removal from the reactor (trip of one, two and more of the reactor coolant pumps, spontaneous closure of isolation valves on the SG, loss of both feedwater supply to the SG and steam extraction from it, loss of power), loss of primary coolant with leakage into a room or from secondary to primary circuit (rupture of pipings inside the guard vessel, breakage of pump nozzles, rupture of tubes and pipings in the steam generator, etc.). Not only single failures in the safety systems were taken into account in the analysis of transients (according to the single failure principle), but also coincident occurrences with great number of failures up to the loss of safety system trains and whole systems. Plant personnel errors were considered as accident initiators and were taken into account in the analysis of post-accident mitigative actions.

Radiological consequences of accidents for the plant personnel and for the environment have been analysed. Radioactive product releases do not exceed the limits established by standards. For the majority of accidents, the radiological consequences are below those caused by the natural radiation background.

Risk reduction

Along with the deterministic validation of the reactor plant safety an assessment of the core damage has been made taking account of sequences of failures resulting in a beginning of core dryout.

The design solutions adopted for VPBER-600 exclude practically a core melt. The probability of a severe core damage is less than 10^{-8} per reactor-year.

External and internal hazards

The reactor plant is designed for external and internal impacts which have to be taken into account in the design of NPPs of the new generation. The external impacts are earthquake, air plane crash, explosive shock wave, wind load, flooding etc. All systems important to safety are designed for maximum design basis earthquake of magnitude 8 to the MSK-64 scale.

The plant containment is designed to withstand an aircraft crash and a shock wave, thus preserving the reactor and the safety-grade systems which are housed in it. Moreover, the containment fulfils the functions of a shadow shield for the safety systems housed in the reactor building annex.

Other external impacts are adopted as standard for the conditions of Russia and can be specified in conformance to site-specific conditions of a particular customer.

Internal impacts are flying (missiles) and dropping objects, fires, and flooding. The design of the reactor plant systems ensure safety under conditions of the internal impacts. For this aim, taking into account the reactor plant features, detailed analyses have been made for such challenges to the guard vessel integrity as breakage of an RCP nozzle followed by blow of the pump on the guard vessel, ruptures of pipings and their whip impacts on the vessel. It was shown that the guard vessel preserves its integrity even in those cases.

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

Safety systems configuration

Figure 6.5-1 gives a principal diagram of the VPBER-600 safety systems. The main features of the safety systems and their interactions in fulfilling the safety functions are described in Section 6.5.6.1.

6.5.6.3 Severe accidents (beyond design basis accidents)

Severe accident mitigation strategy

An estimated frequency of the reactor core damage accident is less than 10^{-8} per reactor·year, what exceeds substantially the requirements of the Russian Code for nuclear safety OPB-88 (10^{-7} per reactor year), as well as the recommendations of IAEA and safety requirements for NPPs established in majority of countries (10^{-6} per reactor·year).

In spite of this feature a special analysis has been performed during the VPBER-600 design development, that resulted in proposals on countermeasures against a severe core damage. The following properties of the reactor plant were taken into account in the analysis conducive to the solution of problems arising in the case of fuel meltdown: low core power density, large surface of the reactor pressure vessel bottom, availability of the guard vessel housing the reactor.

Severe accident prevention and mitigation features

A special system providing discharge of steam-gas mixture from the guard vessel into receiver tanks via a bubbler, has been introduced to mitigate the consequences of a severe accident with a core melt. The corium will be retained on the RPV bottom or, in the case of the RPV melt-through, in a core catcher located in the guard vessel. The bubbler tank and receivers are arranged in a bottom part of the containment. Their total volume does not exceed 1000 m^3, and they do not cause an increase in the overall dimensions of the reactor building.

The analysis performed showed that the pressure and temperature in the guard vessel and in the supplementary tanks during such an accident will only insignificantly exceed the design values; therefore their integrity can be guaranteed. Oxygen is not present in the "reactor-guard-vessel-supplementary tanks" system, and so hydrogen safety is ensured. Releases of radioactive products beyond the plant boundaries will be small and meet the requirements to siting of nuclear district heating plants established in Russia.

After necessary delay for cooling a corium and decay of radionuclides, the gas mixture might be released slowly through a filtering system without hazard to the environment.

6.5.7 Plant layout

6.5.7.1 Buildings and structures, including plot plan

Figure 6.5-7 shows the general layout of the VPBER-600 plant buildings The reactor plant is located within the containment structure, arranged in the reactor building.

The tanks of the emergency heat removal system are arranged in annexes to the reactor building, diametrically opposite to the containment, thus excluding their simultaneous failure at an air crash or shock wave impact.

The containment, the annexes to the reactor building, the reactor, guard vessel, primary circuit equipment and safety systems are designed for an earthquake of magnitude 8 (on the MSK-64 scale).

The reactor plant equipment items layout provides accessibility for performing repair and maintenance operations.

6.5.7.2 Reactor building

The reactor building is shown on Figures 6.5-8 and 6.5-9.

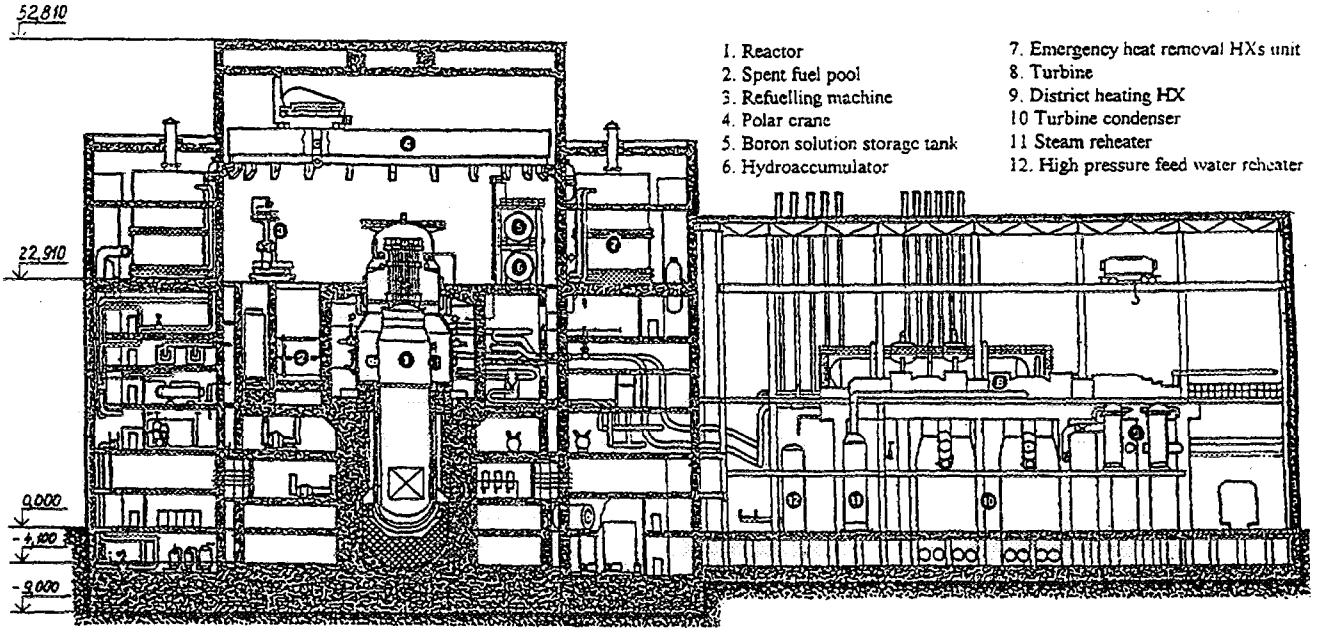


Figure 6.5-7 VPBER-600 - Section through main buildings

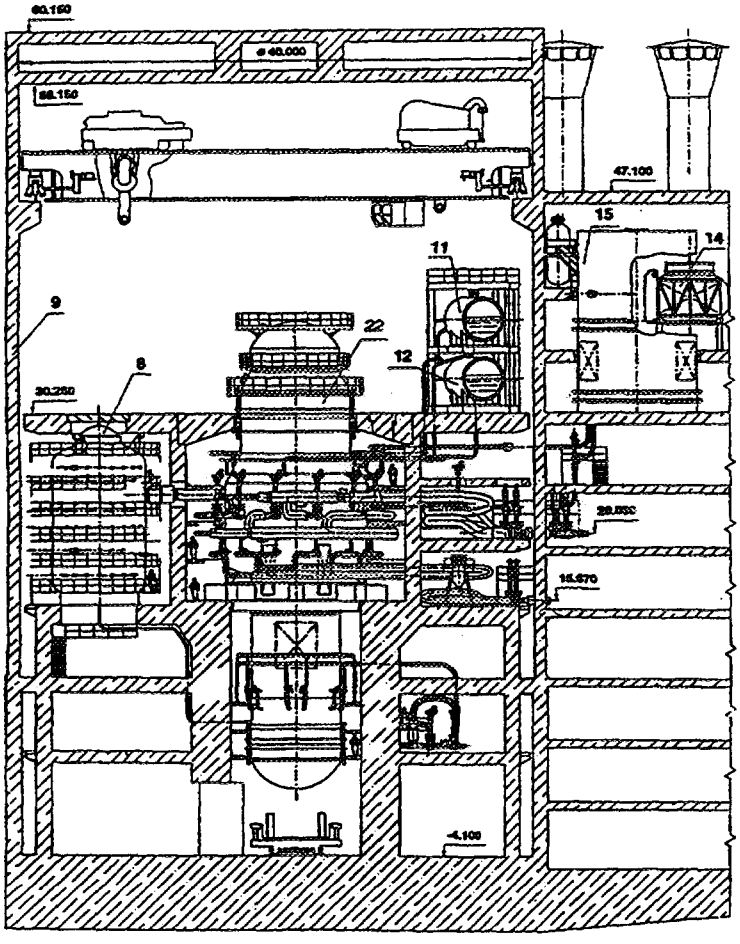


Figure 6.5-8 VPBER-600 - Reactor building section (Table 6.5-I for legend)

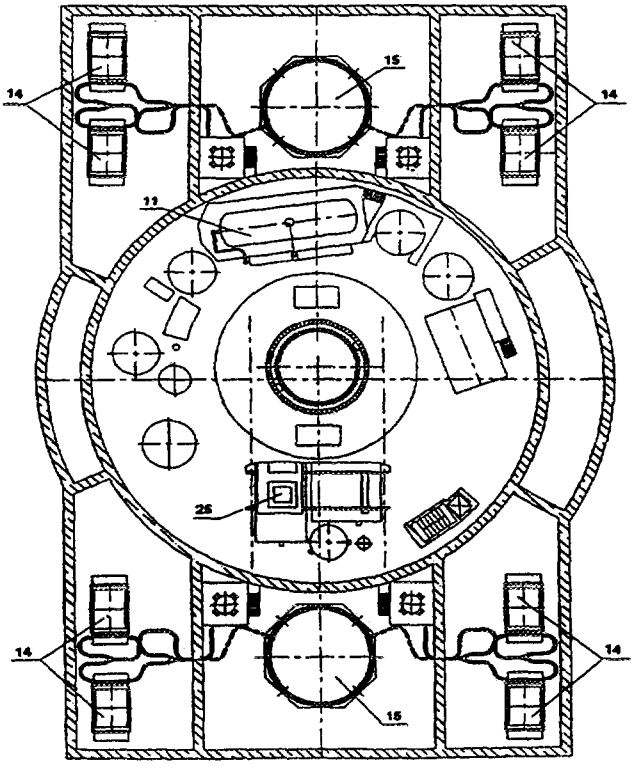


Figure 6.5-9 VPBER-600 - Reactor building, reactor service room level (Table 6.5-I)

6.5.7.3 Containment

Since the reactor and the primary circuit equipment are located inside the guard vessel, a single ferro-concrete containment, without prestressing, but with metallic liner, has been selected. It is designed for an overpressure of 0.1 MPa.

The containment serves for protecting the reactor plant against external impacts, as well as for confining radioactive products at incidents during the reactor refuelling and at accidents with postulated loss of the guard vessel integrity.

The containment serves for protecting the reactor plant against external impacts, as well as for confining radioactive products at incidents during the reactor refuelling and at accidents with postulated loss of the guard vessel integrity.

The containment is a cylindrical structure of 40 m ID with a plane roof. The wall thickness is 0.9 m. The design leak rate from the containment is 0.3% of volume per day.

Corium Retention

Supposed systems for corium retention are described in Section 6.5.6.3.

6.5.7.4 Turbine building

No information provided.

6.5.7.5 Other buildings

No information provided.

6.5.8 Technical data

General plant data		
Power plant output, gross		MWe
Power plant output, net	630	MWe
Reactor thermal output	1 800	MWt
Power plant efficiency, net	35	%
Cooling water temperature		°C
Nuclear steam supply system		
Number of coolant loops	4	
Primary circuit volume, including pressuriser		m^3
Steam flow rate at nominal conditions	950	kg/s
Feedwater flow rate at nominal conditions		kg/s
Steam temperature/pressure	305/6.38	°C/MPa
Feedwater temperature/pressure	230/	°C/MPa
Reactor coolant system		
Primary coolant flow rate	10 140	kg/s
Reactor operating pressure	15.7	MPa
Coolant inlet temperature, at RPV inlet	294.8	°C
Coolant outlet temperature, at RPV outlet	325	°C
Mean temperature rise across core		°C
Reactor core		
Active core height	3.53	m
Equivalent core diameter	3.04	m
Heat transfer surface in the core	4465	m^2
Fuel inventory [UO_2]	62.1	t
Average linear heat rate	10.8	kW/m
Average fuel power density	32.9	kW/kg U
Average core power density (volumetric)	69.4	kW/l
Thermal heat flux, F_q	403.1	kW/m^2
Enthalpy rise, F_H		

Fuel material	Sintered UO_2	
Fuel assembly total length	4 800	mm
Rod array	triangular	
Number of fuel assemblies	151	
Number of fuel rods/assembly	287	
Number of guide tubes for control rods/instr.	18/1	
Number of spacers		
Enrichment (range) of first core	1.0/ 3.6/ 4.0/ 4.4	Wt%
Enrichment of reload fuel at equilibrium core	4.0/ 4.4	Wt%
Operating cycle length (fuel cycle length)	18-24	months
Average discharge burnup of fuel	52 000	MWd/t
Cladding tube material	Zr 4	
Cladding tube wall thickness	0.65	mm
Outer diameter of fuel rods	9.1	mm
Overall weight of assembly		kg
Active length of fuel rods	3530	mm
Burnable absorber, strategy/material	Chromium diboride	
Number of control rods	139	
Absorber rods per control assembly	18	
Absorber material	boron carbide (B_4C)	
Drive mechanism	Magnetic jack	
Positioning rate	4-10	mm/s
Soluble neutron absorber	boric acid solution	
Reactor pressure vessel		
Cylindrical shell inner diameter	5 440	mm
Wall thickness of cylindrical shell	265	mm
Total height	23 960	mm
Base material: cylindrical shell	15×2MPA steel	
RPV head		
liner	08×18H10T steel	
Design pressure/temperature	18/350	MPa/°C
Transport weight (lower part)	880	t
RPV head		t

Steam generators

Type	once-through, vertical	
Number	1, in 12 sections	
Heat transfer surface	13 930	m^2
Number of heat exchanger tubes	66 400	
Tube dimensions	13.1×1.5	mm
Maximum outer diameter		mm
Total height (tube)	3 800	mm
Transport weight of section	25	t
Shell and tube sheet material		
Tube material	titanic alloy	

Reactor coolant pump

Type	Canned, Single-stage, centrifugal pump	
Number	6	
Design pressure/temperature	18.0/350	MPa/°C
Design flow rate (at operating conditions) [8400 m^3/h]	1 690	kg/s
Pump head	5.14	MPa
Power demand at coupling, cold/hot	2 870/2 320	kW
Pump casing material	Stainless steel	
Pump speed	1 500	rpm

Pressuriser

Total volume	80	m^3
Steam volume: full power/zero power		m^3
Operating pressure, total/steam	15.7/11.8	MPa
Design pressure/temperature	18.0/350	MPa/°C
Heating power of the heater rods		kW
Number of heater rods		
Inner diameter	5 440	mm
Total height		mm
Material		
Transport weight		t

Pressuriser relief tank, if any

Not applicable

Total volume		m^3
Design pressure/temperature		MPa/°C
Inner diameter (vessel)		mm
Total height		mm
Material		
Transport weight		t

Guard vessel

Total gross volume m^3	1 440	
Design pressure/temperature	4.0/	MPa/°C
Inner diameter (vessel)		mm
Height/diameter (max)	36 510/11 000	mm
Material	steel	
Design leakage rate	0.15	vol%/24 h

Primary containment

Type	Dry, single wall	
Overall form (spherical/cyl.)	cylindrical, in non-reinforced concrete	
Dimensions (diameter/height)	40/36.51	m
Free volume	60 000	m^3
Design pressure/temperature (DBEs)	200/	kPa/°C
(severe accident situations)	/	kPa/°C
Design leakage rate	0.3	vol%/24 h
Is secondary containment provided?	no	

Reactor auxiliary systems

Reactor water cleanup,	capacity		kg/s
	filter type		
Residual heat removal,	at high pressure		kg/s
	at low pressure		kg/s
Coolant injection,	at high pressure		kg/s
	at low pressure		kg/s

Power supply systems

Main transformer,	rated voltage	3×500/20	kV
	rated capacity	3×260	MVA
Plant transformers,	rated voltage	20/6.3	kV
	rated capacity	2×20	MVA
Start-up transformer	rated voltage	200 (500)/6.3	kV

rated capacity	2×40	MVA
Medium voltage busbars (6 kV or 10 kV)	4×6	kV
Number of low voltage busbar systems	12×0.4	kV
Standby diesel generating units: number	2 + 2	
rated power	2×1.6, 2×0.5	MW
Number of diesel-backed busbar systems	2 + 2, 4	
Voltage level of these	6 + 0.4, 0.4	kV ac
Number of DC distributions	5 + 4	
Voltage level of these	220	V dc
Number of battery-backed busbar systems		
Voltage level of these		V ac

Turbine plant

Number of turbines per reactor	1	
Type of turbine(s)	condensing, K-600-5.9/500	
Number of turbine sections per unit (e.g. HP/LP/LP)	1 HP/ 2LP	
Turbine speed	3000	rpm
Overall length of turbine unit		m
Overall width of turbine unit		m
HP inlet pressure/temperature		MPa/°C

Generator

Type	water-cooled, T3B	
Rated power		MVA
Active power	630	MW
Voltage	20	kV
Frequency	50	Hz
Total generator mass		t
Overall length of generator		m

Condenser

Type		
Number of tubes		
Heat transfer area		m^2
Cooling water flow rate		m^3/s
Cooling water temperature		°C
Condenser pressure		hPa

Condensate pumps

Number	3×50%	
Flow rate		kg/s
Pump head		
Temperature		°C
Pump speed		rpm

Condensate clean-up system

Full flow/part flow		
Filter type		

Feedwater tank

Volume	3	m^3
Pressure/temperature		MPa/°C

Feedwater pumps

Number		
Flow rate		kg/s
Pump head		
Feedwater temperature		°C
Pump speed		rpm

Condensate and feedwater heaters

Number of heating stages	7	
Redundancies		

6.5.9 Project status and planned schedule

The VPBER-600 Nuclear Power Plant is developed by:

- OKB Mechanical Engineering (Nizhny Novgorod): reactor plant project;
- Nizhny Novgorod Institute "Atomenergoproject": project of plant level systems and structures (architect-engineer);
- Scientific-Production Association "Leningrad Metal Works Plant" (St. Peterburg): turbine plant project;
- Scientific-Production Association of automatics and instrumentation (Moscow): project of automated control system;
- Russian Scientific Center "Kurchatov Institute" (Moscow)- scientific management.

The conceptual design of the reactor plant, turbine plant, automated process control system and main principles for the nuclear power plant has been already completed.

A first version of the preliminary safety report, based on the conceptual design of the reactor plant and the main principles of the nuclear power plant project, has been developed and presented to Russian regulatory organizations for review.

The transition to natural coolant circulation in the primary circuit, the reliability and efficiency of the emergency heat removal system are confirmed by experiment and calculations.

Technical-economic assessment made for a nuclear power plant with the VPBER-600 reactor shows that it is competitive with oil-fired power plants: specific cost of its construction in 1991 prices amounts to approximately 1000 roubles per kW of installed power; specific cost of electricity is 0.023 roubles per/kW-hour. This cost of electricity is about equal to that of a natural gas-fired power plant of a similar size and some 1.5-2.0 times less than for a coal-fired power plant.

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CHAPTER 7. OUTLOOK AND EXPECTATIONS FOR THE FUTURE

7.1 Nuclear power projections to 2015

Projecting future nuclear power development is a somewhat difficult exercise, since a number of factors which may influence policies, decision making and implementation of programmes cannot be assessed with certainty.

Up to the year 2000, the installed nuclear capacity world-wide will grow to between 367 GWe and 375 GWe, compared to 340 GWe in 1994 (Figure 7-1), and 344.4 GWe in 1995. Since all the units to be commissioned by the turn of the century are already under construction, the range of uncertainty reflects potential delays in construction and licensing. New nuclear units will be connected to the grid mainly in Asia, while in Western Europe and North America the installed nuclear capacity will remain practically unchanged. In Eastern Europe, although some of the units under construction will be completed, the economic transition will delay significantly the implementation of the nuclear programmes in most countries.

After the turn of the century, the range of uncertainty regarding nuclear power development is wider, owing to a number of technical, economic, environmental and policy factors. The low and high nuclear generating capacity projections developed by the IAEA up to 2015 are based upon a review of nuclear power projects and programmes in Member States. They reflect contrasting but not extreme underlying assumptions on the different driving factors that have an impact on nuclear power deployment. These factors and the ways they might evolve, vary from country to country, and the IAEA projections do not reflect the whole range of possible futures from the lowest to the highest feasible, but provide a plausible range of nuclear capacity growth by region and world-wide.

In the low case scenario, the current barriers to nuclear power deployment are assumed to prevail in most countries during the coming two decades. Economic and electricity demand growth rates remain low in industrialized countries. Public opposition to nuclear power continues, and environmental concerns, such as the risk of global climate change, do not become strong driving factors in energy policies to switch from fossil to nuclear energy. Institutional and financing issues prevent the implementation of previously planned nuclear programmes, in particular in countries in transition and in developing countries, and there is no drastic enhancement regarding nuclear technology adaptation and transfer, nor financial support to developing countries for the implementation of nuclear power projects.

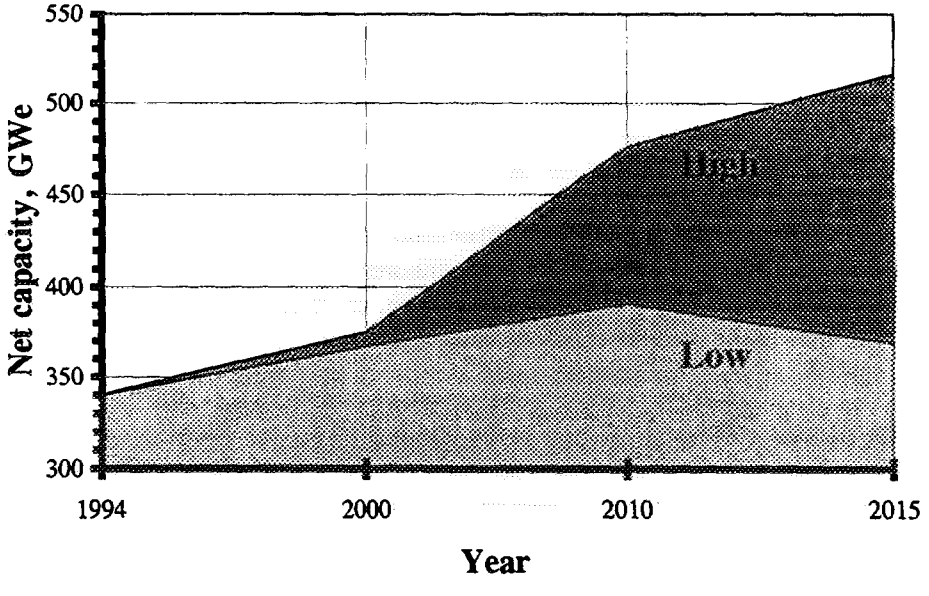


Figure 7-1 World-wide nuclear power outlook up to 2015

Under these rather pessimistic assumptions, most of the nuclear units under construction would be completed but new nuclear units would be ordered only in the countries where nuclear power is a major component of electricity generation mixes, such as France, Japan and the Republic of Korea. Owing to the large number of units that would be shut down at the end of their scheduled operating lifetime, the total nuclear capacity in the world would start to decrease after 2010 and would be similar in 2015 to that in 2000, i.e., at some 370 GWe. The share of nuclear power in the world electricity supply would decrease from about 17% at present to some 13% in 2015.

The high case reflects a moderate revival of nuclear power deployment in an increasing energy demand environment, that might result in particular from a more comprehensive comparative assessment of the different options for electricity generation, integrating economic, social, health and environmental aspects. This case assumes that some policy measures would be taken to facilitate the implementation of these programmes, such as strengthening of international co-operation, enhanced technology adaptation and transfer, and establishment of innovative funding mechanisms. Based on these assumptions, the total installed nuclear capacity world-wide would reach some 515 GWe in 2015 and the share of nuclear power in total electricity generation would be some 15%.

In both the low and high cases, the production capabilities of the world nuclear industry would exceed the demand for new reactors. A higher rate of nuclear power development would be technically feasible and economically viable in a number of countries. However, a substantive revival of nuclear power programmes would require policy measures, including a removal of the de facto moratoria in several countries and the introduction of mechanisms for providing funding support to nuclear projects in developing countries, which seem unlikely to be implemented in the short term.

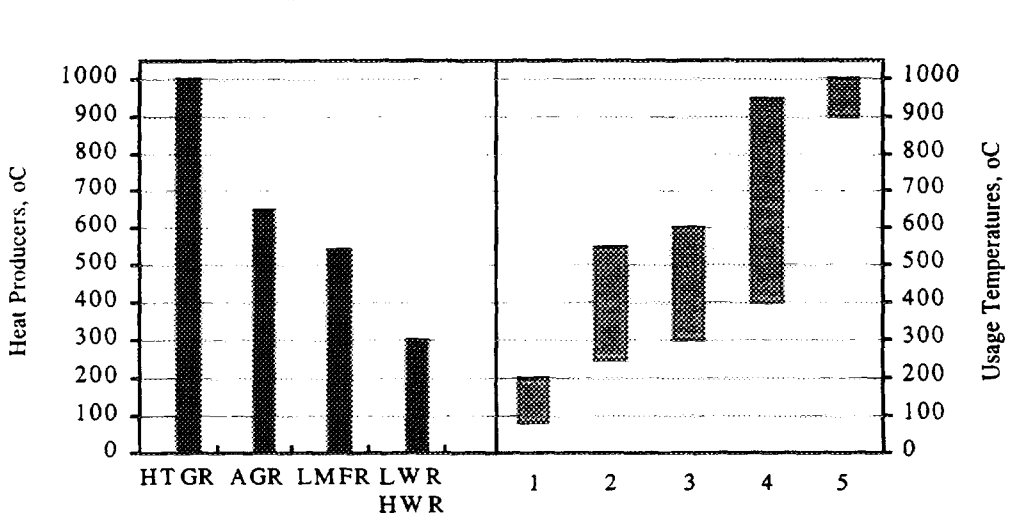
7.2 Non-electric applications of nuclear energy

Today, only a small portion of the nuclear energy produced is used for non-electric applications (a total of only 5 GWth to supply hot water and steam). However, at present, about 30% of the world's primary energy consumption is used for electricity generation, about 15% is used for transportation and the remaining 55% is converted into hot water, steam and heat. This shows that the potential market for applications of nuclear energy in the non-electric energy sector may be quite large. Non-electric applications include desalination, hot water for district heating, heat energy for petroleum refining, for the petrochemical industry and for the conversion of hard coal or lignite. The potential for applications of nuclear energy in the non-electric sector is quite large.

For non-electric applications, the specific temperature requirements vary greatly. Hot water for district heating and heat for seawater desalination require temperatures in the 80 to 200°C range, whereas temperatures in the 250 to 550°C range are required for petroleum refining processes. The use of heat for enhancing heavy oil recovery can be applied by the method of hot water or steam injection. The temperature and pressure conditions required for heavy oil recovery are highly dependent on the geological conditions of the oil field; the temperature requirements range up to 550°C and above.

Oil shale and oil sand processing requires temperatures ranging from 300 to 600°C, and processes used in the petrochemical industry require higher temperatures, in the range of 600 to 880°C. Still higher temperatures (up to 950°C) are needed for refining hard coal or lignite (for example, to produce methanol for transportation fuel), and temperatures of 900 to 1000°C are necessary for the production of hydrogen by water splitting (Figure 7-2).

Water-cooled reactors can provide heat up to about 300°C, and liquid-metal-cooled fast reactors produce heat up to about 540°C. The gas-cooled reactors can provide even higher temperatures, about 650°C for advanced gas-cooled, graphite-moderated reactors (AGRs), and up to 950 to 1000°C for high-temperature gas-cooled reactors (HTGRs).



Legend for user processes:

1 - District Heating, sea water desalination
2 - Petroleum refining
3 - Oil shale & oil sand processing
4 - Refinement of hard coal and lignite
5 - Hydrogen by water splitting

Figure 7-2 Temperature ranges in production and use of nuclear energy

There is considerable incentive to utilize the capability of nuclear plants to provide co-generation of electricity, steam and heat for residential and industrial purposes. Experience in co-generation with water-cooled reactors has been gained in the Russian Federation, Peoples Republic of China, Canada, the Czech and Slovak Republics, Switzerland, Germany, Hungary and Bulgaria. One of the largest uses of nuclear process steam is at the Bruce Nuclear Power Development Facility in Ontario, Canada, where the CANDU PHWRs are capable of producing 6000 MWe of electricity as well as process steam and heat for use by Ontario Hydro and an adjacent industrial energy park.

7.3 Nuclear power development in industrialized countries

In the industrialized countries of the world, the total energy demand and consumption is not expected to rise significantly in the coming decades, but an increase in the demand for electricity supplies is expected.

Energy consumption trends show that electricity plays an important role in modernization efforts and in total energy use efficiency improvements; an increased use of electricity is always the result. It is also obvious that a full participation in the information and communication age, which the world is now entering, will require availability of reliable sources of electricity.

Techno-economic studies have shown that there is a distinct correlation between the trends of electricity consumption and national economic output in a wide range of countries; data shows that electricity consumption continues to grow with increasing gross domestic product GDP, even in countries where the total energy consumption is not rising, or even declining (Figure 7-3). It is also clearly shown that the electricity consumption increase (relative to the GDP increase) is rising rapidly in developing countries.

From 1960 to 1990, the share of electricity in the global energy consumption has grown from 17 to 30% and the annual consumption of electricity per capita has almost tripled (from 765 to 2225 kWh per person), even though two billion people in the world still do not have access to electricity in their homes.

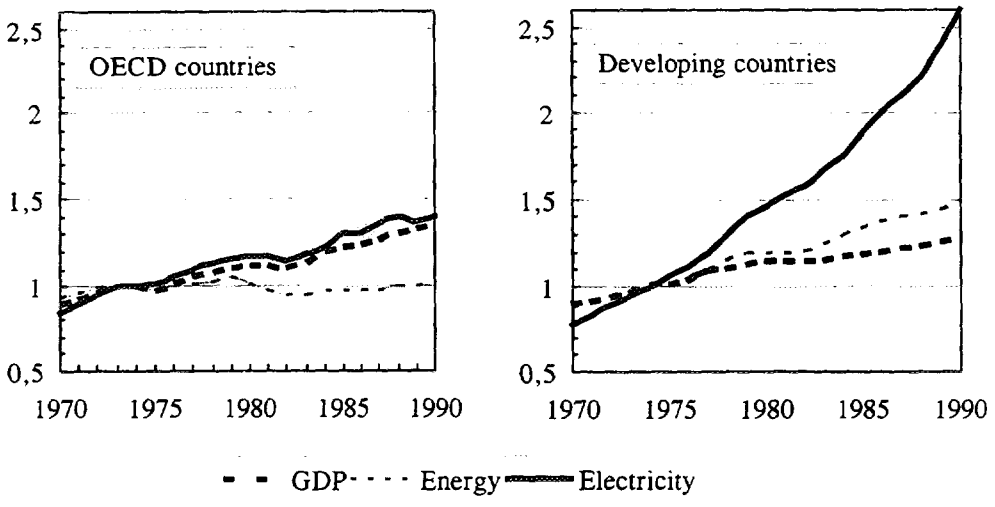


Figure 7-4 Growth trends for Gross Domestic product (GDP), energy and electricity (Per capita values, normalized to 1,0 in 1974)

Looking at the potential market situation in the industrialized countries in the world, it is rapidly concluded that the near-term outlook for new nuclear power plants is rather grim.

In the USA, no nuclear power plants are under construction, and ordering of new plants is not expected before the turn of the century; there is no need for large additional base-load capacity due to a low increase rate of the demand for electricity. In Canada, there is at present a significant surplus of generating capacity, and no plans for building new nuclear plants.

Interconnection of the transmission grid systems, and the development towards free trade and transfer of electricity between countries, in some regions of Europe has provided more planning flexibility for the utilities with respect to operation and utilization of existing power plants and to planning of new capacity; peak load demands can be covered by temporary import from another country where the peak occurs at a different time, and the need for "spinning reserve" and standby capacity decreases. As a result, additional large base load capacity is in many countries not seen as necessary, at least not before the turn of the century. Austria turned down the nuclear option in a referendum in 1975, and has no intention of reversing that decision. Belgium has a large nuclear share of the electricity generation but has no plans on further investments in nuclear. Finland was considering construction of a fifth nuclear unit but the Parliament voted against the project; a significant additional generation capacity is still needed in Finland to meet the projected increase in electricity demand. In France, two more units of the N4 series will remain under construction by the end of 1996, but no further construction starts are envisaged by EdF (Electricité de France) before the turn of the century. Germany has no plants under construction and the German utilities do not believe that new nuclear capacity will be ordered before the turn of the century.

Italy does not have any nuclear power plants in operation, and has had a moratorium on construction of nuclear power plants, following the Chernobyl accident in 1986. Nuclear energy is considered an important option in Italy as a means of building up an indigenous energy source, but a nuclear power plant order within this decade is far from likely. In the Netherlands, the total electricity generation capacity exceeds the needs, and construction of a new nuclear power plant is not considered an option for the future. In Spain and Switzerland there is a moratorium on construction of new nuclear power units, and in Sweden, there is a Parliamentary decision on phase-out of all nuclear power plants in operation by 2010, and a law that forbids utilities to be involved in any planning of new nuclear power plants in Sweden.

In Japan, the situation is different. The Ministry of International Trade and Industry (MITI) has forecasted that the electricity demand will rise significantly during the period up to 2010, and the

installed nuclear capacity is expected to rise by 30 GWe or more, from the current about 40 GWe, in this time period. In a longer time perspective, studies have shown that Japan should need two to four times this nuclear capacity by 2010 in order to revert to the 1990 level of carbon dioxide emissions from the power producing sector.

A review of nuclear power development plans and programmes in industrialized countries is presented in Annex - Nuclear power developments in the world.

7.4 Nuclear power development and developing countries

As noted above, the energy demands in the world is expected to rise significantly in the coming decades. The world population has nearly doubled over the last three decades and will continue to increase. Current estimates forecast that by the year 2020 there will be 8.1 billion people living on this planet, with some 90% or more of the population increase taking place in developing countries.

In these countries, the current electricity consumption per capita, which may be used as an indicator of standard of living, is very low; one or two orders of magnitude lower than in industrialized countries.

Availability of energy is an important prerequisite for socio-economic development in all parts of the world. Depending on the geographic location and the level of industrialization, final energy is consumed in different forms: as electricity, for transport, or as heat. Among the different energy supply alternatives that countries may choose, nuclear power represents one of the few options that provide a means to produce energy in various forms, i.e. as electricity, low and high temperature process heat, and process steam, both economically and under environmentally acceptable conditions. And the environmental concerns about the effects of burning fossil fuels for energy production are stimulating the interest in low-polluting energy sources. A gradually increased deployment of nuclear energy by introducing new, advanced designs into the energy supply systems represents a real option for meeting future energy needs in all parts of the world.

The building of nuclear power plants is capital-intensive, however, and financing has become a major problem in many countries; in particular, this is the case in most developing countries. For countries with an existing infrastructure and established nuclear power programmes which have developed an indigenous manufacturing capability, the situation is not too bad; they need only to import special components, equipment and know-how from abroad, with a corresponding limited spending of foreign currency.

In most developing countries a suitable infrastructure and manufacturing capability is not available, and they generally have a currency that is not convertable; they will therefore have to buy most of the plant equipment from abroad and will depend heavily on loans from foreign banks or institutions. In this context, it may be noted that when constructing the first nuclear power plant a preferable approach appears to be having a "turn-key" delivery, including a technology transfer programme. In that way, the possibilities of making the first plant project a success, actually a matter of paramount importance for the acceptability of a continued nuclear programme, would be the best, and the transfer of technology will enable the country to gradually develop its own capabilities, and successively increase the domestic participation in subsequent nuclear power plant projects.

Some developing countries, such as China, India and the Republic of Korea, have ambitious programmes for deployment of nuclear power, and are also actively pursuing development of own reactor designs.

- China has three nuclear units (of which one is of own design) in operation, and has ordered another two 900 MWe units from abroad. It also plans construction start of a WWER-1000, and has contracted construction of CANDUs with AECL of Canada. The near-term programme includes serial production of a 600 MWe version of the indigenous PWR design that is in operation at Qinshan, and further development of that type, construction of a heating reactor of own design at Daqin, and construction of a high temperature gas-cooled experiment reactor. Taiwan, China has 6 units in operation and has in 1996 ordered two ABWR units from General Electric.

- India has ten plants in operation (of which six are PHWRs of own design), and another four are under construction. Development activities are concentrated to a 500 MWe PHWR design and a 500 MWe fast, sodium-cooled breeder reactor.
- The Republic of Korea has eleven units in operation, all supplied by foreign vendors, but the tenth unit represents the beginning of a new era; the first of a series of plants based on the System 80 of ABB Combustion Engineering for which more and more of engineering and equipment is supplied by Korean companies. Four PWR plants of the series and three PHWR plants are under construction, and a further nine plants are planned for the period up to 2006. The PWRs currently being built in Korea are named Korean Standard Nuclear Power Plants; they represent the results of continuous efforts for nuclear technology in Korea. Development is underway on the next generation reactor (KNGR), based on the System 80+ NSSS design, aiming at start deployment by the turn of the century.

It is also noted that the Democratic People's Republic of Korea has been promised supply of modern LWR plants of Western standard (two Korean Standard Nuclear Power Plants), that Indonesia has rather firm plans on introduction of nuclear power within some years, that Thailand has been discussing such introduction for years, that Iraq had very firm plans on introduction of nuclear power in the late 70s and early 80s, that the Islamic Republic of Iran is trying to get support for completing the Bushehr units, that Pakistan has one 300 MWe PWR under construction and is discussing the possibilities of having a second such unit built, that Morocco is starting a sea water desalination project using nuclear heat, and that Egypt has had, and still has, plans on utilization of nuclear power.

In Eastern Europe, Armenia has restarted one of its two PWRs, and is seeking help to get also the other restarted, or replaced by modern plants, the Czech Republic has xx units in operation and two units under construction, Romania has one PHWR unit in operation and another three units under construction, the Slovak Republic has two units in operation and four units under construction, and Ukraine has yy units in operation and two (to six) units under construction.

In Latin America, Argentina has two nuclear units and Brazil one unit in operation, and they both have one unit under construction; Argentina is also developing own nuclear reactor designs. Mexico has two plants in operation, but no firm plans for further construction; in the early 80s, it had a very ambitious programme for nuclear power development that had to be abandoned for financing reasons. It can also be noted that Venezuela has been discussing utilization of nuclear power plants for heat and electricity generation; the heat is intended for improving the heavy oil extraction along the Orinoco river.

A review of nuclear power development plans and programmes in developing countries is presented in Annex - Nuclear power developments in the world.

Annex

NUCLEAR POWER DEVELOPMENTS IN THE WORLD

The following review of nuclear power programmes in the world has been compiled on the basis of published documents, reports from IAEA studies and information provided to IAEA by Member States and other international organizations.

Argentina

Two nuclear power plants, Atucha-1, a 335 MWe PHWR unit imported from Germany, and Embalse, a 600 MWe PHWR unit imported from Canada, are in operation in Argentina, Atucha-1 since 1974 and Embalse since 1984. The nuclear units amount to less than 10% of the electricity generation capacity of the country but provide some 15% of the electricity supply.

A third nuclear unit, Atucha-2, a 692 MWe PHWR unit of German design, has been under construction since 1981 and is about 75% completed; lack of funding and economic uncertainties have delayed its completion. Besides, the privatization of the electricity sector, in the framework of Argentina's overall privatization programme, has led to extensive restructuring in the nuclear industry and of CNEA (the Comisión Nacional de Energía Atómica), a governmental body previously in charge of the operation of nuclear power plants. The planned sale of the two operating nuclear power plants is expected to provide funds that will be used for bringing Atucha-2 into operation before the end of the 1990s.

Armenia

The Medzamor nuclear power plant, with Armenia -1 and -2, two 376 MWe PWR units of the Russian WWER-440, version 230, design, remained closed in 1994. These units were supplying some 40% of Armenia's electricity demand prior to their shutdown early in 1989 in the wake of an earthquake in the region; the units, connected to the grid in 1976 and 1979, respectively, were not considered adequately designed to withstand the impact of earthquakes.

Armenia continued to experience dramatic energy shortages in 1994, aggravated further by cuts in imported gas and oil supplies. Over the last few years, the Government of Armenia has been pressing for the restart of the nuclear power plant, which is essential to the country's energy supply, and calling for technical and economical assistance for supporting the necessary safety enhancement of the two units before re-commissioning. A number of missions by international experts, including WANO (the World Association of Nuclear Operators) and the IAEA, have provided a comprehensive assessment of necessary safety upgrade measures before a restart of the two units.

Agreements signed early in 1994, under which the Russian Federation will provide technical assistance and nuclear fuel supply to Armenia, represent a major step forward. The overall cost of upgrading the safety of the nuclear units seems to be beyond the financing capabilities of Armenia, however, and would require loans from industrial countries and/or development banks. With technical assistance from the Russian Federation, and economic support from Armenians living abroad, refurbishment has progressed steadily and one of the units might restart by mid-1995, although concern has been expressed about the feasibility of reaching an acceptable level of safety within such a short lead time.

Belarus

Belarus, which has limited domestic energy resources, is facing continuous electricity and heat supply shortages. In 1994, the energy bill for imported fuels was estimated at some US $ 1.5 billion. An energy analysis study for the country has been undertaken with technical assistance from the IAEA, and several energy related projects are being carried out in co-operation with the World Bank and US aid agencies.

A project to build a Soviet design reactor during the 1980s was cancelled after the Chernobyl accident in 1986. The Belarus Government is now considering the implementation of a nuclear programme, aiming at ordering a first nuclear unit before the end of the 1990s. Preliminary discussions on reactor supply have been undertaken with several manufacturers, including Canadian, Russian and US firms. Belarus has also

also expressed interest in resuming the construction of a third nuclear unit in Lithuania, since Belarus is importing electricity from Lithuania.

In preparation for launching a nuclear programme, research and development activities are being carried out, in particular regarding the back end of the fuel cycle, and the education and training of personnel.

Belgium

Belgium has seven nuclear units in operation, providing nearly 56% of the electricity supply of the country. The power plants have a very satisfactory performance, with an average load factor of about 80%.

A governmental decision of December 1988 precludes further investments in nuclear power in the country for the time being. However, the DOEL-3 unit increased its net capacity by some 70 MWe in 1994 with the replacement of its steam generators. Belgian utilities also hold a 25% share in the two French 1400 MWe PWR units under construction at Chooz, close to the Belgian border, which will be commissioned during 1996. The expected low electricity demand growth in Belgium will not require large additional base load capacity and the utilities are building mainly gas turbines for peak load supply. By the end of the century, the share of nuclear power in the total electricity generation is expected to decrease to about 50% (from the current nearly 56%), while the share of fossil-fuelled power plants will increase.

Brazil

Brazil has only one nuclear power plant in operation, Angra-1, a 626 MWe PWR unit of Westinghouse (USA). This unit was first connected to the grid in 1982 and has been in commercial operation since December 1984; it provides less than 1% of the electricity supply of the country. The plant was shut down during more than one year, from March 1993 to June 1994, for extensive maintenance and backfitting.

A second nuclear power plant unit, Angra-2, a 1245 MWe PWR unit of KWU (Germany) design, has been under construction since 1976. The unit is 75% complete but financing problems have delayed completion; connection to the grid is now scheduled for 1998. Angra-3, also a 1245 MWe PWR unit of KWU design, is planned to be in operation in 2004, according to the utility Furnas Centras Elétricas, although funding issues remain to be addressed.

Bulgaria

Bulgaria has six PWR units with a net capacity of 3,538 MWe, in operation at Kozloduy, comprising four 408 MWe WWER-440/230 units (Kozloduy-1 to -4) and two 953 MWe WWER-1000 units (Kozloduy-5 and -6), all imported from the former USSR. Kozloduy-1 and -2 were connected to the grid in 1974 and 1975, the next two units, Kozloduy-3 and -4, in 1980 and 1982, and the two last units, Kozloduy-5 and -6, were connected to the grid in 1987 and 1991; the operation of the last two has been disturbed by a number of problems and the performance records are not very good. However, in spite of extensive outages for maintenance and safety upgrading, the nuclear units of Kozloduy supply more than 35% of the electricity consumption of the country.

Technical assistance and economic support from the European Community, WANO, the IAEA and the European Bank for Reconstruction and Development have helped Bulgaria to upgrade the safety of the nuclear units, enhance the training of operation and maintenance staff, and strengthen the safety culture. Most of the improvements have been carried out with Bulgarian engineering, equipment and staff.

The nuclear power programme of the country includes: continued enhancement of the technical and safety performance of the existing nuclear units until the end of their lifetime; development of policies and facilities for the management and disposal of spent fuel and the decommissioning of nuclear power plants; and building of additional nuclear units. New nuclear units would allow Bulgaria, which has no alternative energy sources, to meet its electricity demand, anticipated to increase with the economic recovery, when the older units at Kozloduy will have to be decommissioned.

A second nuclear power plant project in Bulgaria was the Bellene project on the Danube river, near the Romanian border, originally planned for four to six 1000 MWe WWER-1000 units imported from the former USSR. Construction started in 1986 but stopped in 1990, mainly due to public opposition, with the first unit about 65% completed. The Government is now considering completion of this unit with safety enhancements to meet international standards.

Canada

Canada has 22 nuclear units in operation, representing a total capacity of 15.7 GWe. All these units are PHWRs of domestic design (CANDU PHWR) and 20 of them are operated by the same utility, Ontario Hydro. The share of nuclear power in the total electricity supply of Canada is some 19%; hydro is the major electricity generation source, with a share of some 60%.

Owing to a low economic growth rate, electricity demand is increasing more slowly than had been expected, and there is no need for additional base load generation capacity in the short term. In the Ontario Province, the electricity generation capacity is even exceeding the level required to meet domestic demand. In order to reduce its surplus generation capacity, Ontario Hydro was considering an early retirement of the 848 MWe unit Bruce 2, in commercial operation since 1977. However, an alternative solution is being explored, which would include the refurbishment and adaptaion of Bruce 2 for burning mixed oxide fuel (MOX), allowing the reactor to use plutonium from dismantled warheads. According to this project, another unit, Bruce 1, would also be adapted and used for burning weapons grade plutonium.

There are no plans to build new nuclear power plants but studies are being carried out on extending the life of the units in operation beyond their 30 year design lifetime. In spite of the absence of new orders in the domestic market, at least in the short term, the Canadian nuclear industry continues the development of new reactor designs (e.g., the CANDU-9), focusing for the time being on the export markets. In particular, Atomic Energy of Canada Limited (AECL) signed in 1994 a nuclear co-operation agreement with China; AECL's discussions are focusing on the supply of at least two 700 MWe CANDU-6 units. Canada has also a continuing co-operation with Argentina, Brazil and the Republic of Korea in the field of heavy water reactor fuel design.

China, Peoples Republic of

The first nuclear power plant built in China is the 288 MWe Qinshan, a PWR unit of own design. The operating experience of this unit that was connected to the grid in 1991, is good. The two units of the Daya Bay nuclear power plant, 906 MWe PWR units imported from Western Europe, were connected to the grid in 1993 and 1994, bringing the installed nuclear capacity of China to 2.1 GWe. At present, nuclear power represents hardly more than 1% of the electricity generation capacity of China, which mainly consists of coal fired power plants.

China has expanded its electricity generation capacity at an annual growth rate exceeding 10% during the last decade. Up to year 2000 and beyond, China alone is expected to account for 20% of the increase in world electricity consumption. In order to meet this rapidly increasing demand, China should build additional capacity at a rate of some 10-15 GWe per year during the next two decades. Coal is likely to remain the major electricity generation source in China which enjoys large domestic resources, and there is a potential for a substantial increase of the hydropower capacity since only 10% of the estimated reserves have been exploited. However, nuclear power offers an attractive alternative option for logistic, economic and environmental reasons. About 80% of the coal reserves are concentrated in the north and northwest, and demand for more coal would lead to a large growth in transportation infrastructure, causing an environmental burden and an energy cost increase. Furthermore, coal burning is giving rise to concerns regarding atmospheric pollution, especially in densely populated urban areas.

The ambitious nuclear programme initiated by China aims at alleviating power shortages and environmental burdens. According to the China National Nuclear Corporation (CNNC), commissioning of more than twenty nuclear units, representing some 17 GWe, are planned for the next two decades. A contract has been signed in 1995 (in November) with the French vendor Framatome for the supply of two 900 MWe PWR units to be built at Lingao close to Daya Bay, scheduled for commissioning in 2002 and

2003. Another four to six 1000 MWe units are planned for the Guangdong Province. At Qinshan, two 600 MWe PWR units of Chinese design are under construction and scheduled for commissioning in 2001 and 2002; these units are intended to be the first of a series of standardized units to be built at Qinshan and in other provinces of China. In the Jiangsu Province, the construction of two 1000 MWe units of the WWER-1000 type, imported from the Russian Federation, is planned, and design studies have started in co-operation with the Russian supplier. Agreements have also been signed between the CNNC and the Atomic Energy of Canada Ltd (AECL) for the construction of two 700 MWe PHWR CANDU-6 reactor units at Qinshan. Additional CANDU-6 units could be ordered before the turn of the century.

In Fujian Province, where large additional electricity generation capacity would be needed to meet the demand, the construction of nuclear units is estimated to be a viable option provided that funding mechanisms can be implemented. Nuclear reactors for district and process heat supply are also being developed. A 200 MWth low temperature heating reactor is planned at Daqing City in the north-east of China, and an experimental 10 MW gas-cooled reactor (HTR) is currently under construction.

Croatia

Croatia has no nuclear power plant in operation but shares the electricity generated by Krško, a 632 MWe PWR operating in Slovenia, which supplies some 17% of the electricity consumption of the country. The project to build a nuclear power plant in Croatia, with the output to be shared with Slovenia, is currently on stand-by owing to the prevailing political and economic conditions. Croatia considers this project to be essential to meet the electricity demand of the country, which has limited alternative resources.

Cuba

The nuclear power programme of Cuba was launched in the early 1980s, in co-operation with the former USSR, in order to alleviate the country's dependence on oil imports. The construction of two 408 MWe nuclear units of the WWER-440 type started in 1983 at Juraguá but was suspended in 1992 owing to the end of Russian support and to financial constraints.

The dependence of Cuba on imported energy is very high, and its electricity generation capacity does not match demand. Therefore, completion of the nuclear units is considered essential by the Government. A feasibility study for the achievement of the first unit, which is reported to be more than 75% complete, was carried out in 1994 by a group of consultants from Western Europe and Brazil in co-operation with the Russian Federation and Cuba. On the basis of the results from this study, it is expected that the plant will be completed with technical assistance from the Russian Federation and financial support from foreign investors.

Czech Republic

The Czech Republic has four PWR units with a net capacity of 1,648 MWe in operation at Dukovany, comprising four 412 MWe WWER-440/213 units imported from the former USSR, providing some 30% of the electricity supply of the country. These reactors are of the latest Soviet design and considered to meet international standards. The construction started in 1978 and the four units were connected to the grid in 1985, 1986 (2), and 1987. The average load factor of the four units has been close to 80% since the commissioning in the mid-1980s.

The Czech Republic has established a domestic nuclear industry, which manufactures major components of nuclear power plants, and has a comprehensive education and training programme for highly qualified nuclear engineers and scientists. An independent nuclear safety authority has been established and the Government has taken actions to ensure the management and final disposal of spent fuel, which previously was returned to the former USSR.

Two WWER-1000 units are under construction at Temelin. After assessing the economic and environmental aspects of different alternatives, including conversion of the Temelin units into gas turbine power plants, the Government decided, in March 1993, to complete the construction as nuclear units with modern instrumentation and control technologies that will upgrade their safety to meet international standards. Concerns regarding the need to alleviate atmospheric pollution arising in particular from lignite

fuelled power plants were an important factor in the decision. It was also found that, given current natural gas prices, the cost of nuclear generated electricity in the Czech Republic is around 60% of the cost of gas generated electricity. A contract was signed with Westinghouse (USA) in March 1994 for the supply of instrumentation and control equipment and fuel assemblies for the Temelin units, with funding by a loan from the American Export-Import Bank. The first Temelin unit is expected to be commissioned in 1998.

Egypt

Egypt has been considering the development of a nuclear programme for several years in order to meet its rapidly increasing electricity demand. Pre-feasibility studies have been carried out and one 600 MWe unit might be ordered in the coming years provided that adequate financing mechanisms are implemented.

Finland

Finland has four nuclear units in operation providing about 30% of the electricity generation of the country, which imports more than 60% of its energy supply. The average load factor of the Finnish nuclear power plants is over 85%, among the absolut top performers of the world.

The electricity demand increased by some 7% last year and is projected to grow at a rate of some 5% per annum in the coming years. This will call for an additional generation capacity of some 2-5 GWe by the turn of the century, and the industry was planning for construction of a fifth nuclear unit. In spite of the support from the Finnish Government, industry and labour unions, the project was stopped, however, by a decision of the Parliament in September 1993, saying that additional electricity supply will have to be provided by domestic biomass and peat and from imports. Representatives from the utilities and large energy/electricity consuming industries have requested that the energy policy of the country be reconsidered and the nuclear option be better assessed, taking into account security of supply, economic factors and environmental issues. In the meantime, the power companies have launched programmes to increase the capacities of their plants. Teollisuuden Voima Oy, operating the two 735 MWe BWR units at Olkiluoto, intends to increase their combined power by about 250 MW, and Imatran Voima is aiming towards a nearly 100 MW combined increase for its two VVER-440 units at Loviisa. Thus, the present upgrading programmes would add a total of about 1/3 GW to the nuclear capacity in Finland before the turn of the millenium.

France

In 1994, French nuclear power plants generated some 340 TWh, representing 75% of the total electricity production of the country. The domestic electricity demand increased by less than 1% but France exported more than 60 TWh. The large share of nuclear power in the French electricity supply has allowed electricity prices to remain stable and among the lowest in Europe. It has also permitted drastic reductions in atmospheric emissions of carbon dioxide, sulphur dioxide and nitrogen oxides. Carbon dioxide emissions per capita are 50% lower in France than in Germany or the United Kingdom.

The 1200 MWe demonstration fast breeder reactor, Superphénix, was authorized to restart in 1994, after a four-year outage; it will be used mainly for research and development in the field of plutonium and actinide burning.

The 540 MWe Bugey 1, the last GCR in France, was shut down in May 1994 after 22 years of operation, and the first phase of dismantling, which includes removal of the fuel, was accomplished in 1995. The complete dismantling will be carried out over a period of four to five decades.

A series of four 1450 MWe PWR units of the new N4 design are under construction, at Chooz and Civaux; the first one has been commissioned in 1996 and the other three will follow within two years. The expected growth of the economy and of electricity consumption will not require the ordering of new nuclear units before the turn of the century, taking into account the improvements obtained in the performance of the units in operation, which attained an average availability factor of more than 80% in 1994. The additional peak load capacity required in the coming years wil be provided by gas turbines.

Starting early next century, Electricité de France plans to order a 1450 MWe nuclear unit every one to two years. The EPR, which is being designed and developed jointly by Framatome and Siemens, is scheduled to be commercially available by that time and might be chosen by the French utility for a new series of standard units after the N4 series.

Germany

Nuclear power plants supply more than 29% of the electricity consumption in Germany. The average availability factors of the units in operation are excellent and the cost of nuclear generated electricity is competitive as compared to alternatives, in particular domestic solid fuels. Nuclear power is also contributing significantly to the reduction of atmospheric emissions from electricity generation. In 1994, around 150 million tonnes of CO_2 emissions, i.e., more than 15% of the total emissions in Germany, were avoided by the operation of the nuclear power plants. Therefore, the Federal Government considers that nuclear power should remain a significant component of the electricity supply mix of Germany.

However, the nuclear power development programme has been frozen for several years mainly due to lack of socialand political consensus on nuclear policy issues. During recent years, several nuclear units have been shut down for extended periods of time, pending decisions from the safety and licensing authorities of different Federal states. In the light of the potential economic impacts of policy uncertainties, utilities are reluctant to embark on nuclear projects. Moreover, the low growth rate of electricity demand, which has remained below 1% per year during the last decade, does not call for large additional base load capacity, if any.

In spite of the views expressed by the Federal Government that nuclear power will have to contribute substantially to electricity generation for reasons of security of supply, economic competitiveness and environmental protection, new nuclear units are unlikely to be ordered before the turn of the century. On the other hand, a total phase-out of nuclear power within 15-30 years is considered desirable by some political parties. In this context, nuclear R&D efforts focus on the enhancement of reactor safety and the back end of the fuel cycle, including final disposal of radioactive waste.

Hungary

Hungary has four PWR units with a net capacity of 1,729 MWe in operation at Paks, comprising four 430 MWe WWER-440/213 units imported from the former USSR. Paks-1 and -2 were connected to the grid in 1982 and 1984, and Paks-3 and -4 in 1986, and 1987. The performance records of the Paks units are very impressing. These units that are of the later WWER-440 version and meet international standards, have benefited from the Finnish experience on enhancement of Soviet design reactors. Moreover, a programme on maintenance, training and safety enhancement is being carried out in Hungary with financial support from the European Union's PHARE project and from the US Government, and with technical assistance from the IAEA.

Nuclear power supplies more than 40% of the electricity consumption of Hungary and is a major contributor to the security of energy supply in the country. In the context of transition to a market economy, emphasis is given to energy efficient improvements and there is no immediate need for additional base load electricity generation capacity. Furthermore, one of the objectives of the energy policy is to reduce the energy dependence on the Russian Federation. Therefore, the project to implement two 1000 MWe WWER-1000 reactors at Paks was cancelled in 1989.

India

India has ten nuclear units in operation with a total capacity of 1,695 MWe and four units under construction. The operating nuclear units comprise two 150 MWe BWR units at Tarapur of General Electric (GE) design imported from the USA, two PHWR units of 90 and 187 MWe, respectively, at Rajasthan, imported from Canada, two 155 MWe PHWR units of domestic design at Kalpakkam, and two pairs of 202 MWe PHWR units of domestic design, at Narora and Kakrapar. The Tarapur units were connected to the grid in 1969, and the Rajasthan units in 1972 and 1980. The construction of the domestic Kalpakkam units began in 1971 and they were connected to the grid in 1983 and 1985. The Narora units were connected to the grid in 1989 and 1992, and the Kakrapar units in 1992 and 1995.

The nuclear power plants supply only some 3% of the electricity consumption in India; the major portion, more than 70%, is generated in coal-fired power plants, and hydropower plants supply some 25% of the total electricity generation. The demand for energy and electricity increases dramatically in India due to rapid population growth and economic development. The electricity consumption is growing at a rate of about 8% per year, and, despite a steady growth of electricity generation capacity in recent years, the gap between demand and supply remains high; the shortage exceeds 10 GWe in capacity and 20 TWh in electric energy. Still, the electricity consumption per capita is less than 300 kWh, among the lowest in the world.

The units under construction are 202 MWe PHWR units of domestic design expected to be commissioned before the end of the century, between 1997 and 1999. The completion of the units under construction, initially planned for 1996, has been delayed by the failure of the concrete containment dome of the Kaiga-1 unit and by financial difficulties.

When India initiated a nuclear power programme in the early 1960s, it aimed at diversification of energy supply and the implementation of a domestic industry. At present, the concerns on diversification and security of supply are supplemented by environmental concerns as the driving factors for nuclear power development, but technical and financial difficulties as well as constraints on technology transfer have enforced a drastical reduction in the mid-term nuclear power programme; construction of a number of 450 MWe PHWR units of the domestic design, as well as imports of 1000 MWe PWR units, is still planned for the next two decades.

Indonesia

The Government and the national atomic energy agency of Indonesia (BATAN) have been considering the implementation of a nuclear programme for more than ten years as part of the electricity system expansion plan, which reflects an electricity demand growth rate of some 12 to 15% per annum. In the island of Java, some 11 GWe of additional capacity would be needed before the end of the century to meet the expected increase in electricity demand. Although Indonesia has large oil resources, diversification of supply is a key objective of the energy policy of the country. Avoiding drastic increase in domestic oil consumption is essential for supporting the economic growth, which is largely dependent on the revenues from oil export. Moreover, the oil reserves of the country will be exhausted in the coming decades and Indonesia will become a net importer of oil early in the next century. Although alternative energy resources (coal, gas and renewable sources) are available, nuclear power is considered necessary as part of the electricity generation mix.

The present nuclear programme of BATAN calls for the construction of twelve units with a total capacity of 7 GWe, which would be commissioned over a period of twenty years starting just after the turn of the century. The total cost of the programme is estimated to be some US $ 17 billion, and financing mechanisms would have to be established between Indonesia and foreign investors or development banks in order to support its implementation. A major milestone in the Indonesian nuclear programme was achieved in 1994 with the completion of the first phase of a feasibility study carried out for the Indonesian Government by a Japanese consultant NEWJEC. The study, which concludes that nuclear power could be economically competitive with alternative options, considered various 600 MWe and 900 MWe reactor types, and provided the basis for selecting a site for the first nuclear power plant on the northern cost of Java. Bidding procedures are being prepared and, according to BATAN, tenders for the first unit will be requested in 1995 and construction could start by the end of the century. However, besides financing issues, concerns regarding the safety of nuclear power plants built in a densely populated seismic region might delay implementation of the nuclear programme.

Islamic Republic of Iran

The Islamic Republic of Iran is a large producer and exporter of oil, but the electricity generation capacity of the country is some 15% lower than required to meet the demand. In the mid-1970s, a major nuclear power programme was underway, and construction of two nuclear power plants, two 1200 MWe PWR units imported from Germany, started at Bushehr in 1975. In 1979, this nuclear power programme was suspended, and construction activities stopped, at a fairly advanced stage of the civil work for the two units.

The Islamic Republic of Iran resumed a nuclear power programme in 1991 with a bilateral agreement with China for the supply of two 300 MWe PWR units of Chinese design, similar to the Qinshan power plant. The agreement was confirmed in 1993 even though the schedule for construction has not yet been fixed. In 1994, the Ministry of Atomic Energy of the Russian Federation (MINATOM) and the Atomic Energy Organization of Iran (AEOI) agreed on a scope of work for completing one of the units at Bushehr with a 1000 MWe PWR unit of the WWER-1000 type. The Russian design reactor will be constructed using the infrastructure already in place and could be completed within a few years, according to MINATOM, once the financing issues are settled. The supply by the Russian Federation of another 1000 MWe unit and two 440 MWe units of the WWER type is being discussed.

Italy

Italy imports some 15% of its electricity and a large share of this import is provided by French nuclear power plants. Although the moratorium on nuclear power expired in 1992, there has been no decision taken regarding the revival of the country's nuclear programme. However, Italy remains active in the field of nuclear R&D, with activities related to advanced reactor designs and safety enhancements. Italy is also participating in the European programme to provide technical support to Eastern Europe countries for enhancing the safety of Soviet design reactors. The engineering company Ansaldo is engaged in a number of projects outside Italy, on feasibility studies and technical support for the construction and refurbishment of nuclear power plants.

Japan

In 1994, the Ministry of International Trade and Industry (MITI) of Japan released its "long term energy supply-demand outlook" covering the period up to 2010. Security of supply remains a driving factor of the Japanese energy policy, which is also aiming towards economic competitiveness and environmental protection. In particular, the MITI outlook reflects the greenhouse gas emission reduction targets set up by the Framework Convention on Climate Change (FCCC). As compared to the previous MITI outlook published in 1990, the total energy demand is projected to grow at a slightly lower rate owing to a reduced economic growth rate and ambitious targets for energy efficiency improvements. However, the electricity demand growth rate is projected to be higher than in the previous outlook, and the total electricity demand is expected to exceed 1100 TWh in 2010 compared to 800 TWh in 1993.

With regard to energy and electricity supply, the 1994 outlook shows some drastic changes reflecting the actual trends experienced during the last four years. Coal consumption has been revised downward, in the light of the environmental protection objectives, but the share of coal-fired power plants in total electricity generation remains the same as in the previous outlook, i.e., at 15% in 2010. Natural gas consumption has increased sharply in Japan during the last four years, and large capacities for importing liquid natural gas (LNG) are now in place. This trend is therefore expected to continue, even though reducing natural gas imports would enhance the security of supply and alleviate the economic risk associated with market price volatility. One of the major changes relates to the penetration of renewable energy sources, which has been revised drasically downwards. This revision reflects the fact that the economic and technical prospects for intermittent renewable sources, as well as biomass, have not improved recently in Japan inspite of steady R&D efforts.

The prospects for nuclear power development in Japan are among the most impressive in the world. The nuclear installed capacity is expected to increase from 39.9 GWe in 1995 to 70 GWe or more in 2010, and the nuclear share in total electricity generation is expected to grow from some 31% to 42%. In the long term, Japan will need a nuclear installed capacity of 160-300 GWe in order to revert to the 1990 level of carbon dioxide emissions by 2010, according to a study from the Institute of Energy Economics of Japan.

The competitiveness of nuclear generated electricity as compared to fossil-fueled alternatives, i.e., coal, oil and LNG including combined cycles, is well established in Japan. The costs of electricity generated by renewable energy sources, i.e., geothermal, photovoltaic, wind and biomass power plants and fuel cells, are estimated to be substantially higher than the cost of nuclear generated electricity.

By the end of 1995, Japan had 51 nuclear units in operation, representing a total installed capacity of nearly 40 GWe. Three units were connected to the grid in 1994, the 280 MWe FBR Monju, the 825 MWe BWR Onagawa-2, and the 890 MWe PWR Ikata-3. The construction time for Ikata-3 was seven years and four months; for the Onagawa unit it amounted to five years and four months. Three units with a total capacity of 3.8 GWe were under construction, - the 1180 MWe PWR Genkai-4, and the two 1356 MWe BWR units (of ABWR type) no. 6 and 7 at Kashiwazaki Kariwa. Kashiwazaki Kariwa-6 was connected to the grid in January 1996, after a construction period of three years and two months; unit No. 7 will follow early 1997. Genkai-4 was connected to the grid in November 1996, after a construction period of four years and four months.

Although no planned reactors have received governmental approval, some ten new orders from Japanese utilities are expected in the near future and connections to the grid are scheduled to continue at a rate of one nuclear unit per year during the enxt decade.

Japan has a large and diversified programme for research and development in the field of advanced reactors. Following the commissioning of the Monju prototype, it is planned to design and construct a demonstration FBR. Research on high temperature gas-cooled reactors is also pursued by the Japan Atomic Energy Research Institute (JAERI). A 30 MWth high temperature test reactor (HTTR) under construction at Oarai.

Kazakstan

The BN-350, a 70 MWe liquid metal cooled reactor, is the only nuclear unit in operation in Kazakstan. The construction started in 1964, and it was connected to the grid in 1973. The lifetime load factor amounts to 50.5% and the energy availability factor to 87.5%; the unit is, in addition to electricity generation, used for potable water production (10,000 t/d) and process heat supply. If used for electricity production alone, its output can be increased up to 130 MWe.

The industry in the country is highly energy intensive resulting in a large demand for electricity and heat; most of the electricity (some 80%) is generated by coal fired power plants. The nuclear industry that was developed when the country was part of the USSR, includes uranium production, fuel cycle service supply and research facilities. In particular, the reserves of uranium in Kazakhstan are estimated to exceed 400 000 t U and its annual production capability is over 3 000 t U. The large uranium resources, together with the availability of qualified manpower and existing industrial infrastructure, offer favourable conditions for the development of a nuclear power programme. Moreover, the construction of nuclear units would help meeting the expected electricity demand growth without increasing atmospheric emissions resulting from coal burning.

The nuclear programme under consideration includes the replacement of the BN-350 by an advanced liquid metal cooled reactor and the construction of one or two small size heat and power supply reactors and two to four 600 MWe WWER-640 units. These units of Russian design are tentatively planned for commissioning between 2005 and 2015.

Republic of Korea

The Republic of Korea has an economic growth rate of some 10% per year and expects a rapid increase in energy demand. Nuclear power is the major domestic source of energy since the country has practically no fossil fuel reserves. The nuclear share of the total electricity generation is over 40%. The electricity consumption per capita, some 3 MWh per year, is rather low compared to the level of industrial countries, and electricity demand is therefore expected to grow rapidly during the next few decades. The nuclear power programme of the Republic of Korea aims at meeting the increase in demand and alleviating the dependence on imported fossil fuels as well as the environmental burdens arising from fossil fuel burning.

The Republic of Korea has eleven nuclear units in operation with a net capacity of 9 170 MWe, comprising one PHWR unit and ten PWR units. The PHWR unit, the Wolsung Unit 1, is a 629 MWe CANDU-6 reactor unit imported from Canada, with grid connection in 1982. The first unit built in Korea is

the Kori Unit 1, a 556 MWe PWR unit of Westinghouse design, imported from the USA, with grid connection in 1977. Kori Unit 2 is a 605 MWe PWR unit of Westinghouse design that was commissioned in 1983. It was followed by Kori Units 3 and 4, two 895 MWe PWR units of Westinghouse design with grid connection in 1985. The next pair of units, Yongwang Units 1 and 2, two 900 MWe PWR units of Westinghouse design were connected to the grid in 1986. They were followed by the Ulchin Units 1 and 2, two 920 MWe PWR unit of Framatome design, imported from France; their grid connection took place in 1988 and 1989, respectively. The tenth and eleventh nuclear units, the Yongwang Unit 3 and 4, are 950 MWe PWR units of Combustion Engineering design, imported from the USA that were connected to the grid in 1994 and 1995, respectively. The latter units form the basis for the Korean Standard Nuclear Power Plant design.

Seven more units are under construction, four 950 MWe PWR units of the Korean Standard Nuclear Power Plant design, and three 650 MWe CANDU-6 reactors, all of which scheduled for commissioning before the end of the century.

The nuclear power development programme of the Republic of Korea started in the early 1970s and the first units were turnkey imports. Progressively, through technology adaptation and transfer, and national R&D, a comprehensive programme was implemented, leading to the establishment of a domestic nuclear industry. The national utility, the Korean Electric Power Corporation (KEPCO), has demonstrated impressive efficiency in the operation and maintenance of its nuclear units; in 1993 and 1994, the average availability factor exceeded 87%, and the excellent project management, in particular during construction, has promoted the competitiveness of nuclear generated electricity, which is some 20% cheaper than the other sources of generation used in the country's electricity system.

The nuclear R&D programme of the Republic of Korea, carried out by KEPCO and the Korean Atomic Energy Research Institute (KAERI) in co-operation with foreign companies, covers advanced reactor development and fuel cycle activities, including fabrication of enriched uranium and mixed oxide fuel and the management of spent fuel and radioactive waste. The development of an advanced reactor of domestic design is planned to be completed by the end of the century and the first unit of this type is scheduled for commissioning before 2010. The Republic of Korea has also entered the international market as a supplier of nuclear technology, and in 1994 Korean companies signed agreements with China and Turkey.

Lithuania, Republic of

The Republic of Lithuania has two LWGR (or RBMK) units with a net capacity of 2 370 MWe in operation in Ignalina; the two units originally rated at 1500 MWe has been downrated to 1250 MWe for safety reasons. The units that were connected to the grid in 1983 and 1987, are supplying more than 85% of the electricity consumption of Lithuania and allow export of electricity to neighbouring Belarus.

Lithuania receives technical assistance and financial support from the European Union, the European Bank for Reconstruction and Development and the Swedish safety authority SKI for a comprehensive safety enhancement programme. The Lithuanian Government has made significant progress in setting up a legal and institutional framework for the operation, management and safety control of its nuclear power plants. Measures have also been taken to adjust electricity prices towards full cost recovery, including safety upgrades and waste management.

The construction of a third 1500 MWe RBMK unit at Ignalina was stopped in 1989 and is unlikely to resume until the safety enhancement programme on units -1 and -2 is completed, i.e., by the turn of the century. Although the commissioning of this third unit might facilitate the electricity supply in the Baltic States and Belarus, its competitiveness with gas fired power plants would need to be thoroughly assessed before the project is revived.

Mexico

Two nuclear power plants, Laguna Verde-1 and -2, two 654 MWe BWR units of GE (General Electric) design imported from the USA, are in operation in Mexico; Laguna Verde-1 since 1989 and

Laguna Verde-2, since November 1994. The construction work on these units started in 1976 and 1977, respectively but completion was delayed by a series of disturbances and changes of Architect Engineering companies; the work on the second unit was suspended completely between 1982 and 1989. Laguna Verde-1 which supplies some 3% of the total electricity consumption of Mexico, has operated quite satisfactory. The operation of Laguna Verde-2 is expected to raise the nuclear share of the total electricity supply to some 5%.

Population increase and economic development are leading to drastic increases of energy and electricity demand; nearly 80% of the country's primary energy consumption is supplied by hydrocarbons. The energy policy is aiming towards diversification of supply, in particular alleviating the dependence on oil, and reducing atmospheric emissions arising from burning of fossil fuels. The continuation of the nuclear programme could address these two issues, but has been hampered mainly by financial difficulties.

Netherlands

Two nuclear units, with a total capacity of 504 MWe, are in operation in the Netherlands, supplying some 5% of the electricity consumed in the country. The total electricity generation capacity of the Netherlands exceeds that needed to meet the demand.

In 1994, the Dutch Parliament voted, by 77 votes against 73, to shut down by the end of 2003, the 449 MWe PWR unit at Borssele, which was connected to the grid in 1973. This vote led to the cancelling of the license given to the utility, SEP, by the previous government for backfitting of the nuclear unit and continuing its operation until at least 2007. SEP, which had already started a programme for upgrade and life extension at Borssele, will receive monetary compensation from the Government and will proceed with safety-related backfitting of the unit. The 55 MWe BWR unit at Dodeward since 1968 is also undergoing major upgrading.

The Dutch Government has prepared a nuclear energy white paper, which was presented to the Parliament in 1995.

Pakistan

The Kanupp nuclear unit in operation in Pakistan, a 125 MWe PHWR unit imported from Canada, generated nearly 0.6 TWh in 1994, with an availability factor exceeding 85%. Over the lifetime, however, the capacity and energy availability factor is 26.8%. The nuclear power provides only some 1% of the electricity supply of the country, which is facing drastic power shortages.

A second nuclear unit, under construction at Chasma, is scheduled to be connected to the grid by the end of 1998. This unit, Chasnupp-1, a 300 MWe PWR unit imported from China, is an improved version of the Qinshan unit put into operation in China in 1993. Discussions on the supply of a second Qinshan type unit by China have been initiated.

Philippines

The only nuclear unit in the Philippines, a 600 MWe PWR imported from the USA, was mothballed in 1986 owing to political changes in the country. Continuing legal disputes with the American manufacturer are blocking decisions on either the completion of this unit or its conversion to a fossil fired plant.

The country has limited domestic energy resources and is highly dependent on imports for its supply. Energy demand growth is expected to exceed 7% per year up to the end of the century. The electricity generation capacity is already not sufficient to meet present demand and drastic shortages are experienced. The power development plan calls for the addition of some 20 GWe to the installed capacity by 2005.

In this context, the government is considering the revival of a nuclear programme and has recommended to undertake a siting and feasibility study for the construction of several nuclear units.

Poland

The construction at Zarnowiec of two WWER-440 units from the former USSR stopped in 1990 owing to financial difficulties and concerns regarding the safety of Soviet design reactors. All the equipment

that had been acquired by Poland for the construction of these units has been sold. The present energy policy of the country imposes a moratorium on nuclear power development until the end of the century. Coal is the major energy resource and fossil fuel burning is leading to atmospheric emissions that exceed the Western European standards. The Governement is considering resuming a nuclear programme, relying reactors of advanced design that would lead to commissioning of a first 600 MWe unit by 2010.

Romania

Romania has one 630 MWe PHWR unit imported from Canada in operation, and another four 650 MWe PHWR units under construction at Cernavoda. Construction of the five units started in 1980, 1982, 1984, 1985, and 1986, respectively, and the first unit was connected to the grid mid-1995, after some slight delays due to lack of funding and difficulties in acquiring equipment. After commissioning, this unit supplies some 8% of the electricity consumption of the country.

The electric utility (RENEL) is discussing with foreign investors a financial plan for finishing Cernavoda 2, which is around 35% complete. Although the electricity generation capacity exceeds the demand, Romania is planning to complete the nuclear units under construction for exporting electricity to neighbouring countries, thereby improving its balance of payments.

Russian Federation

Nuclear plays an important role in the electricity generation mix of the Russian Federation. Twenty-nine nuclear units of domestic design are in operation, representing a capacity of 19.8 GWe and supplying more than 11% of the electricity consumption of the country.

The ongoing economic transition has resulted in a significant slowing down of industrialized activities, and energy production and consumption have been continuously declining since 1991; the total electricity generation decreased by some 8% from 1993 to 1994. It is estimated that the recovery from the economic crisis will take at least one decade, and the 1990 level of activity will not be reached again before 2005-2010.

The electricity generated by nuclear power plants has been decreasing drastically since 1992, due to extensive outages of some nuclear units for safety upgrading, but also to financing difficulties encountered by the operators. Setting electricity tariffs based on full generation cost recovery would be a prerequisite for ensuring the economic sustainability of the nuclear units as well as of the other power plants in the Russian Federation.

In spite of the economic situation of the country, the Ministry of Atomic Energy of the Russian Federation (MINATOM) still plans to expand the installed nuclear capacity by 15 GWe within the next 15 years. MINATOM's nuclear programme has been approved by the Government but its timely implementation would require addressing financial issues in the very near future.

Four nuclear units under construction are scheduled for commissioning by the turn of the century. These include three 950 MWe WWER-1000 units, the Balakovo 5, Kalinin 3 and Rostov 1 units, and one 925 MWe RBMK unit, Kursk 5. In addition, two 750 MWe liquid metal reactor units are under construction at South Urals and are expected to be in operation by 2005.

Finally, two small units for heat and power supply, with a power capacity of 30 MWe each, are under construction at Voronezh.

Aprogramme of safety upgrading at the WWER and RBMK reactor units of older Soviet design is being pursued with technical assistance from foreign companies and financial support from the European Union and development banks.

The nuclear power development programme of the Russian Federation covers design and engineering studies on various reactor types, including WWER-640, advanced WWER-1000, liquid metasl cooled reactors and small reactors for heat and power supply, district heating and potable water production.

Slovak Republic

Four nuclear power plant units with a total capacity of 1 632 MWe are in operation in Slovakia at Bohunice, supplying close to 50% of the electricity consumption of the country. Two of the Bohunice units are 408 MWe units of the WWER-440/230 type, the older type of the Soviet designed PWR; the two other units are 408 MWe units of the WWER-440/213 type, the most recent design. The units were connected to the grid in 1978, 1980, 1984, and 1985, respectively.

The two first units have been brought up to an international standard level in an extensive upgrading programme carried out from 1991 to 1993, and they are expected to be operated until the turn of the century. The other two are of the more modern design that basically meets international standards, but a programme for further safety enhancement of these plants has also been undertaken by the Slovak Electricity Company (SEP, privatized in 1994) which operates the Bohunice plant, in co-operation with western European companies, following the recommendations of the Slovak safety authority and the IAEA. This programme will be completed in 1997.

Four 388 MWe WWER-440/213 units are also under construction at Mochovce, two since 1983 and two since 1985. The transition to a market economy is expected to lead to a higher efficiency in energy production and use, but Slovakia still needs additional electricity generation capacity and the domestic energy resources - mainly low quality solid fossil fuels - are scarce. A study by the European Bank for Reconstruction and Development (EBRD) concluded that completing Mochovce-1 and -2 represents the least cost electricity system expansion plan for Slovakia. The EBRD was prepared to finance the completion project sponsored by SEP and the French utility Electricité de France (EdF), including upgrading to modern safety standards and completion of the two units, but the opposition to the project by neighbouring countries, in particular Austria, managed to block the EBRD support. In spite of this drawback, Slovakia and SEP have decided to go ahead with the completion project, with technical and financial support from Russia.

Slovenia, Republic of

Slovenia has one 632 MWe PWR unit of Westinghouse design, imported from the USA, in operation at Krško. The construction of this plant started in 1975, and it was connected to the grid in 1981. Its lifetime load factor amounts to 74.3%, and its energy availability factor to 77.4%. Krško supplies about 38% of the electricity consumption in Slovenia but it also supplies electricity to neighbouring Croatia. The operators are planning to replace the steam generators of the unit, aiming at extending the lifetime to forty years or more.

Slovenia has adopted a moratorium on building new nuclear power plants until the end of the century, and the country has no plans to resume a nuclear power development programme.

South Africa

South Africa has two 921 MWe PWR units, imported from France, in operation at Koeberg, supplying almost 6% of the electricity consumption of the country.

There is no immediate need for additional electricity generation capacity, although the demand is likely to increase more rapidly than in the past, owing to social development and urbanization. At present, electricity generated by coal-fired plants and imported hydroelectricity are the least cost options in South Africa. The nuclear power plants in operation have proved to be safe and reliable, however, and the public opinion is very positive about nuclear energy. Uranium extracted as a by-product from gold mines could support a broad deployment of nuclear power, and South Africa might be ordering additional nuclear units by the turn of the century.

Spain

Seven PWR units and two BWR units are in operation in Spain, with a total capacity of 7.4 GWe. The nuclear units are supplying some 35% of the electricity consumption of the country.

The moratorium on construction of new nuclear power plants, adopted in 1983, resulted in the final cancellation of five units which were at different stages of completion. These units include two 900 MWe PWR units under construction at Lemoniz, which were some 95 and 55% completed, two 975 MWe BWR

units at Valdecaballeros, which were half completed, and one 1000 MWe PWR at Trillo, on which construction work had just started. The mothballed nuclear units will be replaced, according to the energy plan of Spain, by coal-fired power plants using domestic coal, gas-fired power plants using gas imported from Algeria, and electricity imported from France. The extra financial charge incurred by the utilities as a consequence of the moratorium is expected to lead to an increase of some 3.5% in the electricity prices paid by the customers.

In spite of the above, the Spanish nuclear industry continues to participate in research and development activities in co-operation with foreign companies, focusing on enhanced nuclear safety, life time extension and advanced reactors.

Sweden

Sweden has 12 nuclear units in operation, representing a total capacity of 10 GWe. In 1995, the electricity generated by the nuclear power plants amounted to 66.7 TWh, supplying some 47% of the electricity consumption (143 TWh) of the country. The previous year, the nuclear share of the electricity generation (137 TWh) was some 51% with a production of 70 TWh.

In 1980, Sweden voted by referendum to shut down all nuclear units when certain conditions were fulfilled, namely that economically and environmentally acceptable replacements were at hand. In the same year, the Parliament decided that the shutdown of the nuclear units should be made by the year 2010, on the basis of its interpretation of the lifetime of the newest units (25 years). No action has yet been taken regarding the closure of the nuclear power plants nor their replacements, however. Technically and from the economic viewpoint, the lifetime of the nuclear units in operation could be extended to at least 40 years, leading to their shutdown between 2012 and 2025. The cost of nuclear generated electricity is some 65% of the cost of gas generated electricity. Owing to its climate and highly energy intensive industrial structure, Sweden has an annual electricity consumption per capita (nearly 18 MWh) that is among the highest in the world. Since the environmental protection policy of Sweden prevents implementing new large hydro power plants, the alternatives to nuclear power are energy conservation, gas-fired power plants, renewable energy sources and imported electricity. Utilities, large electricity consuming companies and labour unions have requested the Government to reconsider the decision taken in 1980. The parliamentary decision that nuclear power production should end by 2010 still remains valid, however. A parliamentary commission was appointed to review the situation and provide advice to the Parliament. The commission presented its findings at the end of 1995, concluding that it would be feasible to phase out nuclear, but a phase-out to 2010 would be very costly; a phase-out over an extended period was recommended. The present Government has stated that it would like to effectuate the shutdown order for the first unit during this mandate period, and in this context, the commission declared that one unit could possibly be phased out without significant risks for the energy supply security in Sweden (based on considerations of annual energy demand [in TWh] and energy supply capacity rather than "minute-by-minute" balances between power demand [in GW] and power generation).

A new parliamentary decision is due in 1997, and the Government has stated that it wants a decision with a broad majority of the Parliament to ensure that the energy policy will last. Some politicians have argued that the matter of the future of the utilization of nuclear power should be referred to a new referendum, since the current knowledge base is much wider, with respect to both nuclear power and renewable energy sources, and, last but not least, 15 years after the referendum in 1980, one million of the voters then are now replaced by young peple that did not have a say in 1980, - and in 2010, there would be another million voter shift. Most politicians, however, turn down the idea of a new referendum, noting that if the issue is too complicated for the politicians to decide on, it would definitely be far too complicated for the public; that was really one of the lessons learned from the referendum in 1980.

On 4 February, the Government announced an agreement - between the Social democratic party, the Centre party, and the Communists - on initiating a nuclear phase-out with the stop of one of the Barsebeck units on 1 July 1998, with the second unit to follow in 2001, if "replacement" power would be available; a parliamentary decision could be anticipated in June 1997. The costs and implications of the "agreement" are not yet quite clear; the politicians claim that in the longer term, bio mass and wind power will make up for

the capacity loss, and in the mean time new natural gas fired power plants, and import of electricity from Denmark, produced in coal-fired plants without filters and flue gas cleaning, are pointed out as the alternatives. The electricity consuming industry has been promised subsidized electricity prices, and so it is quite obvious that the private consumers will be the ones that will have to pay for the experiment.

Switzerland

The five nuclear units in operation in Switzerland contributed almost 39% of the electricity generation of the country. In December 1994, BeznauII a 350 MWe PWR unit in operation since 1971, was issued a license for operation until the end of 2004. Similarly, the 355 MWe BWR in operation at Mühleberg was issued a ten-year operation license after refurbishment and 10% capacity upgrade in 1992. In both cases, the operating utilities intend to seek an extension of the licenses before they expire. The three other nuclear units in operation have unlimited operating licenses.

In September 1990, Switzerland voted by referendum to impose a ten-year nuclear moratorium which prevents any plan to undertake the construction of a new nuclear unit before the turn of the century.

Taiwan, China

Taiwan, China has six nuclear units with a net capacity of 4 890 MWe in operation. The nuclear units comprise two 890 MWe PWR units at Maanshan, two 604 MWe BWR units at Chin Shan and two 951 MWe BWR units at Kuosheng, all imported from the USA. The PWR units are of Westinghouse design and the BWRs of General Electric design. The Chin Shan units went into operation in 1978 and 1979, the Kuosheng units in 1981 and 1983, and the Maanshan units in 1984 and 1985. The nuclear units were operated at an average load factor of 78.1% during 1994, providing 31.7% of the electricity supply.

Taiwan has been planning for construction of further nuclear units for some time; a project was suspended for political and economic reasons in the late 1980s after the successful bidder had been selected, but now a new project is underway. After careful evaluation, the ABWR of General Electric was found to be the most attractive bid, and a contract was signed in the Summer of 1996.

Thailand

The rapid increase of electricity demand, some 14% per year, calls for around 1 GWe additional electricity generation capacity per year during the next two decades. The government and the national utility, Electricity Generating Authority of Thailand (EGAT), have considered the implementation of a nuclear power programme since the late 1970s. Thailand has a nuclear research centre and studies have been carried out on the potential role of nuclear power in the electricity generation mix as well as on siting of nuclear power plants.

Turkey

Turkey is facing a rapid increase in energy demand and there are environmental concerns raised by atmospheric pollution arising from domestic lignite and coal burning. Since the 1970s, Turkey has been considering the implementation of a nuclear power programme in order to ensure a secure and clean electricity supply, and a site was selected at Akkuyu on the Mediterranean coast. However, negotiations with different reactor suppliers have not resulted in agreement during the last two decades, owing to financial and political issues.

At the end of 1994, the Turkish Electricity Generation and Transmission Company (TEAS) signed an agreement with the Korean Atomic Energy Research Institute (KAERI) regarding the assessment of the Akkuyu site, the review of commercial reactor types available on the market and assistance in bid evaluation. The present nuclear plan of the national utility schedules the commissioning of at least 2 GWe nuclear capacity by 2010, and by the end of 1996 Turkey has again issued a request for bids.

Ukraine

Ukraine has 15 nuclear units with a net capacity of 12 679 MWe in operation on five sites. Three of these are LWGR (or RBMK) units located at Chernobyl. Construction on the 721 MWe units Chernobyl-1 and -2 started in 1972 and 1973, and on the 925 MWe Chernobyl-3 in 1977; they were connected to the

grid in 1977, 1978, and 1981, respectively. Further data can be found in the table below. [It can be noted that the Chernobyl-2 unit has been out of service most of the time since 1991, following a fire in the turbine hall.]

The other nuclear units are two 406 MWe WWER-440 units in operation at Rovno and ten 950 MWe WWER-1000 units in operation at Khmelnitski, Rovno, South Ukraine and Zaporozhe. Rovno-1 and -2 were connected to the grid in 1980 and 1981; South Ukraine-1, -2 and -3 in 1982, 1985 and 1989; Khmelnitski-1 and Rovno-3 in 1987 and 1986, and the five Zaporozhe units were connected to the grid in 1984, 1985, 1986, 1987 and 1989.

More than one third of the electricity (34.2%) consumed in Ukraine is generated by nuclear power plants, and in view of the key role of nuclear power in the country's electricity supply the Government is reluctant to follow the recommendations of the international community to shut down the Chernobyl units, which are causing safety concerns. In spite of the offers made by foreign countries and international organizations to provide technical assistance and financial support for closing down the units, Ukraine is considering keeping them in operation until 2007-2011, when they would reach the end of their lifetime.

The moratorium on the construction of new nuclear units, imposed by the Soviet government in 1990, was abolished by the Ukrainian authorities in 1993, and at least three of the six 950 MWe WWER-1000 units under construction are now scheduled to be put into operation before the end of the century. Zaporozhe-6, which is nearly complete, could be commissioned in 1995. However, the present economic situation in Ukraine is likely to make it difficult to finance the continuation of the nuclear programme without support from industrialized countries and development banks, which in turn will rely on the agreement of an action plan for closing down the Chernobyl units.

United Kingdom

Thirty-five nuclear units are in operation in the United Kingdom, representing a total capacity of 12.9 GWe and supplying almost 26% of the electricity generated in the country. The first PWR unit built in the country, Sizewell B, was connected to the grid in February 1995.

Sizewell B is a modern PWR, built locally under license from Westinghouse, which incorporates a number of enhanced safety features to meet more stringent safety standards. The unit was completed on schedule and within the provisional budget. All the other nuclear units in operation in the U.K. are gas-cooled reactors of Magnox and advanced gas-cooled reactor (AGR) types. Substantial improvements in the performance of the AGRs have been obtained during recent years.

As of April 1995, the national safety authority, the Nuclear Installations Inspectorate (NII), has authorized lifetime extension for all Magnox units which had reached 30 years of operation. Similar lifetime extension authorizations are expected to be obtained for the next Magnox units to be considered. Although the authorizations from the NII do not specify the duration of the lifetime extension, the British regulators have indicated that no safety factors have been identified which would limit the operation of the Magnox reactors to less than 40 years.

The future development of the nuclear power programme will depend mainly of the outcome of the Government's nuclear review, which covers issues related to the privatization of the nuclear industry and the economic viability of implementing new nuclear power plants. Environmental and social issues will also play a role on whether or not to revive a nuclear power development programme. Since international decisions regarding carbon dioxide emission reduction have been postponed, atmospheric pollution concerns might be considered a lower priority in the British energy policy, taking into account the social impacts of reducing domestic coal production. However, the utilities that are operating nuclear power plants, Nuclear Electric and Scottish Electric Ltd, are planning to order new nuclear units for replacing the Magnox and AGRs at the end of their lifetime, in order to maintain the share of nuclear power in the electricity generation of the country at its present level. Nuclear Electric has already submitted, for government review, a project to build two units similar to Sizewell B on the same site, for commissioning in the first

decade of the next century. Nuclear Electric has also carried out preliminary assessment studies on the construction of one PWR unit similar to Sizewell B at Hinkley Point.

United States of America

The USA is the world's largest nuclear electricity producer with some 670 TWh per year, i.e., 30% of the total nuclear electricty generated worldwide. There are 110 nuclear units in operation in the USA with a total net capacity of 100.6 GWe, supplying some 22% of the electricity consumed in the country. The average availability factor of the nuclear power plants has improved significantly since 1990 and has now reached over 78%. The reduced outage rates resulted from improved management and from streamlining of the safety control procedures. The improved performance of nuclear units and a reduction in operating and maintenance costs has led to a reduction in the total cost of nuclear electricity generation.

The nuclear power development programme has, however, been more or less on standby since the 1970s. The units connected to the grid in 1990 and 1993, the Commanche Peak 1 and 2, had been under construction since 1974, and the last unit, Watts Bar 1, a 1165 PWR, under construction since 1972, was connected to the grid in early 1996.

Five units still under construction were definitely cancelled in 1994. In May, the Washington Public Power Supply System (WPPSS), stopped the construction of two PWR units, WNP-1 and WNP-2, representing a total capacity of 2.5 GWe. In December, the Tennessee Valley Authority decided not to complete Watts Bar 2, a 1165 MWe PWR unit, and the two 1260 MWe PWR units at Bellafonte; these units may possibly be converted to gas-fired power plants. In this context, on the assumption that the lifetime of the nuclear units in operation would not be extended, and that no new unit would be ordered by the turn of the century, the installed nuclear capacity and the share of nuclear power in the elctricity supply will decrease after 2005.

The main obstacles to a revival of the nuclear power development programme in the USA are the lack of demand for base load electricity generation, and the economic and radioactive waste disposal issues. The growth in electricity demand, which is not expected to exceed 1.5% per year in the coming 10-15 years, could be met by an increasing supply from non-utility power producers and additional gas turbines mainly for peak load generation. The economic performance of nuclear power plants has been affected by regulatory issues which resulted in greatly extended construction times. Streamlining of the licensing procedures and improved management are likely to enhance the competitiveness of nuclear power plants. Cheap domestic fossil fuels will, however, remain the lowest cost option in a number of regions of the USA. For radioactive waste disposal, the implementation of a final repository for spent fuel is a prerequisite for any utility to consider ordering a new nuclear unit.

In spite of the stagnation of the domestic nuclear programme, the US nuclear industry has maintained its activities in the fields of reactor maintenance and service supply, the export of equipment, and the development of advanced reactors for the domestic and international markets. Four advanced reactor designs have been submitted to the US Nuclear Regulatory Commission (NRC) for approval. Two received final design approvals from the NRC in July 1994, the 1350 MWe PWR System 80+ developed by ABB Combustion Engineering and the 1350 MWe advanced BWR (ABWR) developed by General Electric. The 600 MWe advanced PWR AP-600 developed by Westinghouse is still under review by the NRC; up to mid-1996, 600 MWe simplified BWR (SBWR) developed by General Electric was also under review, but then General Electric stopped work on the 600 MWe version and shifted its emphasis to a unit with larger output.

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Consultants Meetings

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Advisory Group Meeting

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